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EXAMINATION OF T-111 CLAD URANIUM NITRIDE FUEL PINS IRRADIATED UP TO 13 000 HOURS AT A CLAD TEMPERATURE OF 990° C

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16. Abstract

The examination of 27 fuel pins irradiated for up to 13 000 hours at 990° C (1815° F) is described. The fuel-pin clad was a tantalum alloy (T-111) with uranium nitride as the nuclear fuel. Two nominal fuel-pin diameters were tested with a maximum burnup of 2.34 atom percent. Twenty-two fuel pins were tested for fission gas leaks; thirteen pins leaked. Clad ductility tests indicated clad embrittlement. The embrittlement is attributed to hydrogen from an n, p reaction in the fuel. Fuel swelling was burnup dependent, and the amount of fission gas release was low, generally less than 0.5 percent. No incompatibilities between fuel, liner, and clad were in evidence.

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SUMMARY

The examination of 27 fuel pins irradiated for up to 13 000 hours at 990° C (1815° F) in the Plum Brook reactor facility is described. The fuel-pin clad was a tungsten lined tantalum alloy (Ta-8W-2Hf) with uranium mononitride as the nuclear fuel. Two nominal diameters of fuel pins were tested: a 1.905-centimeter (0.750-in.) diameter reference pin and a 0.952 centimeter (0.375-in.) half-diameter pin. The reference-diameter pin was run at a reference burnup rate and the half-diameter pins were run at burnup rates accelerated by factors of either 2 or 4. The half-diameter pins were scaled down from the reference diameter such that the fuel area fraction and the clad area fraction were approximately the same for both sizes. Also thinner clad was used on both the reference and half-diameter pins to accelerate clad strain. The fuel length was 5.72 centimeters (2.25 in.) except for one prototype fuel pin whose fuel length was 38.1 centimeters (15 in.). All fuel pins were instrumented with fuel center thermocouples. Most of the thermocouples lasted the duration of the tests, up to 13 000 hours, at 1125° C (2060° F) and a 4.3×10^{20} nvt fast fluence (where nvt is neutrons/cm^2) without failure.

Curtailment of NASA sponsored nuclear programs prevented a complete examination of all pins. Nevertheless, the results include, although not for all 27 pins, fuel pin integrity, clad ductility, clad strain, fission gas release, fuel burnup, fuel swelling, neutron fluence, metallography, and instrumentation reliability.

Twenty-two fuel pins were tested for fission gas leaks; 13 pins leaked with two having visible cracks. Expansion ring ductility tests were conducted on clad specimens from some of the fuel pins, and all but two pins failed in the range of 1 to 3 percent clad strain, which indicates an embrittlement of the clad. With two exceptions, ductility was restored to the clad by vacuum heat treating the specimens for 1 hour at 1038° C (1900° F). The irradiated clad embrittlement is attributed to hydrogen from the n, p reaction with the fuel.

Fuel swelling appeared to be primarily burnup dependent for burnup values up to 2 atom percent. Maximum burnup reached in these tests was 2.34 atom percent. In general, the fuel pellets did not crack or crumble and fuel metallography remained unchanged.

The amount of fission gas released from the fuel was low, generally less than 0.5 percent. The maximum amount released was 2 percent. No incompatibilities between fuel, liner, and clad were in evidence.

INTRODUCTION

A technology program based on a fast spectrum liquid-metal-cooled compact reactor concept for space power application was conducted at the NASA-Lewis Research Center. The investigation was keyed to a reference reactor design concept with a reactor power level of 1 megawatt (thermal), a coolant outlet temperature of 950° C (1742° F), and an operating life of 50 000 hours. Three reports (refs. 1 to 3) describe the overall effort. The conceptual reactor design is described in reference 1. The nuclear design and associated initial experiments are discussed in reference 2. And the materials technology and fuel pin test programs are discussed in reference 3. The fuel-pin irradiation program discussed in reference 3 was conducted in two reactor facilities: the Plum Brook Reactor Facility (PBRF) located in Sandusky, Ohio, and the Oak Ridge Reactor (ORR) located in Oak Ridge, Tennessee. The test results discussed herein cover the fuel pins irradiated in the Plum Brook Reactor Facility. A previous report (ref. 4) discusses the results of three fuel pins irradiated for 8070 hours. The test results from the pins irradiated in the ORR were obtained from an unpublished report by K. J. Bowles and R. E. Gluyas of Lewis.

The Plum Brook program consisted of irradiating 29 test pins using the T-111 alloy (Ta-8W-2Hf) as the clad and cored UN (uranium nitride, 94 percent of theoretical density) as the fuel. A thin tungsten liner was used to separate the fuel from the clad. The goal of the reference fuel pin design concept is to limit diametral clad strain to 1 percent in 50 000 hours at a clad temperature of 990° C (1815° F) and a burnup of 2.73 atom percent.

Because of the impracticality of inpile testing for 50 000 hours, irradiations were conducted for shorter time periods but included tests at accelerated conditions. These accelerated tests incorporate higher burnup rates and/or thinner clad.

Two fuel pin diameters were tested: a reference diameter pin run at the design burnup rate and half-diameter pins run at either two or four times the design burnup rate.

The purpose of the Plum Brook tests was to evaluate the fuel-pin design under conditions of temperature and total burnup of the reference reactor fuel pins. The fuel-pin evaluation was made primarily in terms of dimensional stability and the ability of the clad to contain both fuel and fission products. In addition, the test results were to be used to provide data to improve modeling techniques for fuel swelling and clad strain. However, before this program could be completed, the nuclear space-power reactor program at Lewis was terminated. Nevertheless, an attempt was made to obtain as much information as possible to advance the technology in the time allowed for postirradiation examination. This report includes the results of all the postirradiation examinations performed on 27 of 29 test pins irradiated in the Plum Brook Reactor.

The results include fuel pin integrity, clad ductility, clad strain, fission gas release, fuel burnup, fuel swelling, density measurements, neutron fluence, metallography, and instrumentation reliability. However, all 27 pins were not examined for each of the investigative areas.

Data were taken in the U.S. Customary system of units and converted to SI for reporting purposes.

TEST PROGRAM AND TEST CONDITIONS

The objective of the Plum Brook test program was to evaluate the fuel-pin design in terms of dimensional stability and the ability of the clad to contain both fuel and fission products. The fuel pin design was based on reference design conditions of 50 000-hour life, 2.73-atom percent fuel burnup, and less than 1-percent maximum diametral clad strain at a clad temperature of 990° C (1815° F).

A 50 000-hour irradiation test is impractical, requiring approximately 8 years of in-pile testing when taking into consideration down-time of the test reactor. Consequently, a 20 000-hour test at the reference burnup rate was selected, and accelerated tests were included to allow for extrapolation to 50 000-hour conditions with the aid of modeling (e.g., ref. 5). Accelerated tests (increased burnup rates), however, require smaller fuel diameters to limit the temperature drop across the fuel. Therefore, the reference diameter pin was run at the reference burnup rate and half-diameter pins were run at two and four times the reference burnup rate. The half-diameter pins were scaled down from the reference diameter such that the fuel area fraction and the clad area fraction were approximately the same for both sizes. In addition, a clad area fraction less than the reference value was included on both the reference- and half-diameter pins. Again, the half-diameter thin clad was half that of the thin clad-reference diameter pin. The result was two pin sizes, a reference diameter and a half diameter (scaled down), three burnup rates, and two clad thicknesses (regular clad and thin clad, noting that both are scaled down for the half-diameter pins). The fuel length in each fuel pin was 5.72 centimeters (2.25 in.) except for one full length prototype fuel pin. The fuel length in this pin was 38.1 centimeters (15 in.). All other variables were kept to a minimum in order to facilitate the interpretation of the test results; for example, UN fuel (94 percent theoretical density), T-111 clad, clad temperature [990° C (1815° F), except for two pins run at the off-design condition of 870° C (1600° F)], constant fuel area fraction, and two clad area fractions. The test program variables are listed in table I, and the test specimens to investigate these variables are itemized in table II.

The fuel pins of this test program were irradiated in the PBRF. The physical loca-

tion of these pins relative to the reactor core is shown schematically in figure 1. Also included in the figure are the pin designations.

In order to achieve useful irradiation data, it was important to provide an environment conducive to uniform power generation in the short fuel pins. An attempt to achieve uniform power was undertaken by

(1) Using low enrichment fuel (~5 percent for the reference diameter pin and ~8 percent for the half-diameter pin) to minimize the radial self-shielding that occurs when testing in a thermal reactor

(2) Using hafnium shielding to minimize the circumferential flux variation across the pins

(3) Using axial hafnium shielding to reduce the axial flux variation in the pins.

Reference 4 gives a typical example of how an attempt was made to approach uniform power generation in three test pins irradiated in the same reflector test hole.

FUEL PIN AND CAPSULE ASSEMBLY

The overall fuel pin and capsule assembly design, which is essentially the same for both diameter pins, is shown in figure 2. A brief description of the important design features are presented in reference 4. For further information reference 6 should be consulted. References 7 and 8 describe the full length prototype fuel pin and capsule assembly.

High quality assurance was essential in the procurement of the test pins and capsules. Detailed documentation was kept of all procedures and inspections used during fabrication and assembly of all fuel pins and capsules.

All welding and assembly procedures were fully qualified. High cleanliness standards were maintained from the time components were cleaned until the components were assembled into an instrumented capsule assembly.

Reference 6 includes typical detailed procedure for fuel pin and capsule fabrication, weld qualification, inspection, and assembly. Photographs showing the components and instrumentation of a typical fuel pin and capsule are shown in figures 3 and 4. Thermocouples penetrated the capsule end caps and extended into the thermocouple wells, which were integral with the fuel pin end caps. The thermocouple junctions of all pins with 5.72-centimeters (2.25 in.) long fuel were approximately 3.81 centimeters (1.5 in.) apart.

A stagnant gas gap was used as the method of temperature reduction from the fuel pin clad to the flowing reactor cooling water surrounding the capsule.

The schematic layout of figure 1 shows that three test pins were irradiated in the reflector test holes and that two test pins were irradiated in the lattice test holes. In

these test positions vertical capsule adjustment (to compensate for test reactor control rod withdrawal) was made using an electromechanical device described in reference 9. Provision was made to irradiate up to five full length pins in the horizontal through hole facility. Each full length pin was to be individually controlled to provide adjustment relative to the core. Several full length pins and capsule assemblies were fabricated as described in references 7 and 8. However, only one full length prototype fuel pin was irradiated and that for 1990 hours.

RESULTS AND DISCUSSION

Results were obtained from postirradiation examination of 27 fuel pins and capsule assemblies. The results include information on fuel pin integrity, clad ductility, clad strain, fission gas release, fuel burnup, fuel swelling, density measurements, neutron fluence, metallography, and instrumentation reliability. However, due to the program termination, all 27 pins were not examined for each of these investigative areas. In order to facilitate the postirradiation discussion, two tables are presented. Table III lists a summary of fuel pin operating temperatures and the number of thermal excursions experienced by each pin. Table IV presents a summary of results from the postirradiation examination of the pins.

A thermal cycle has been arbitrarily defined as any decrease in operating temperature in excess of 300° C (540° F). Most of these cycles result from normal withdrawal and insertion during reactor operation required for refueling. However, some of these cycles are a result of a reactor scram, or setback, where the thermal cycles are more severe. The top and bottom fuel center thermocouple readings were recorded every hour, and the average of these readings was used as the fuel pin average centerline temperature as listed in table III. The average clad temperature is less than the average fuel center temperature by an amount that is dependent on the fuel-pin dimensions and the fuel power density.

For the three different burnup rates (power density variation in the fuel) the clad temperature at the beginning of life is calculated to be less than the average fuel center thermocouple reading by the amount listed in table V.

Table IV lists 27 of the 29 irradiated test pins. The two pins not listed failed, one because water leaked into the capsule after 9400 hours of irradiation and the other because a defective fuel pin end cap allowed air to enter the fuel pin prior to irradiation. (Subsequent alterations in inspection procedures precluded the start of irradiation of a leaking pin.) The pin with the defective end cap was removed after 880 hours of irradiation.

Real irradiation times for the other 27 pins ranged from 1150 hours to 13 000 hours.

Accelerated burnup rate tests reached the equivalent of 35 400 hours of burnup (2.34 at. %) in the reference design reactor.

Fuel Pin Integrity

Twenty-two fuel pins were tested for fission gas leaks - nine regular clad and thirteen thin clad. Three of the nine pins with regular clad leaked, and ten of the thirteen pins with thin clad leaked. Although thirteen total fuel pins leaked, only two had visible clad cracks. One of the pins with a visible crack is shown in figure 5. Two others had leaks located in the end-cap thermocouple well region. More specifically the leaks were found in the end cap containing the closure weld. In one case the leak location was in the thermocouple well. The results of a liquid bubble type of leak test indicated a leak emanating from the base of the thermocouple well viewed from the fuel side of the end cap. The thermocouple well is an integral part of the end cap. Two smaller leaks were also observed in the face of the end cap (side also facing the fuel) in the vicinity of the seal weld closure. The seal weld was also leak tested. No leaks were detected in the top of the seal weld (side not exposed to the fuel). A photomicrograph of a metallographic section taken through the top end cap (thermocouple well wall and closure weld with filler wire) is shown in figure 6. The approximate magnification is ×40. Note the leak path from the fuel side of the closure hole through the wall of the thermocouple well. The location of the other nine leaks in the pins was not determined. However, it is believed that the other leaks may also be located in the closure-weld end-cap region. This conjecture is based on the following:

(1) No cracks were visually observed in the clad.

(2) Hydrides are more stable at the lower temperatures in the end caps, which enhance their susceptibility to embrittlement in this region.

(3) The combination of higher temperature gradients in the end-cap region and the end-cap geometry containing a closure weld and thermocouple well produce higher thermal stresses.

(4) Microcracks resulting from machining the thermocouple well could lead to crack initiation in the end cap.

The cracks of the two fuel pins containing visible cracks coincided with the seam of the tungsten liner. The seam produced a step of at least 0.025 millimeter (1 mil) as shown in figure 7. The step in the liner would cause higher local stresses. Using the final pin diameter measurements from the 503I pin, (one of the pins with a visible crack) the clad strain as listed in table IV is 1.36 percent. However, the visible clad crack was about 0.508 millimeter (20 mil) wide. A calculation was made of the magnitude of the clad strain before the crack opened. The calculated clad strain (crack closed) was

about 0.6 percent. (A technique was later developed whereby the tungsten liner was bonded directly to the clad. The resulting seam was less pronounced. This method was used for the full length fuel pins.)

Figure 8, which shows the fuel stack removed from fuel pin 505E, further illustrates the effect of the seam. Note the axial impression that resulted from the fuel contacting the seam of the tungsten liner. This pin did not have a visible crack. It was irradiated for 7000 hours real time (equivalent to 28 000 hr of reference burnup). The leak was isolated to the end-cap region. A summary of the test pin results are as follows:

Six pins were accelerated by a factor of two so as to attain burnup rates equivalent to burnup times ranging from 23 000 to 26 000 hours. Three pins leaked and three did not. Four of the six pins were regular clad. The three pins that did not leak were regular clad.

Eight fuel pins were accelerated by a factor of four so as to attain burnup rates equivalent to burnup times ranging from 28 000 to 35 400 hours. Seven of these pins leaked. Six of the eight pins were thin clad. Of the two regular clad pins one did not leak.

Four reference diameter pins were checked for leaks. Three of these pins did not leak, one of which was irradiated for 13 000 hours. Axial cracks in two pins were the result of a seam causing a stress concentration. The absence of the seam would not necessarily have prevented a pin failure.

It is believed that most failures were located in the thermocouple-well closure weld region.

Clad Ductility Tests

Expansion-ring ductility tests were conducted on clad specimens from some of the fuel pins. Control tests were also conducted on unirradiated ring specimens. These control test specimens were from aged [2862 hr at 1900° (1038° C) in a vacuum] T-111 material and as-fabricated fuel pins. The clad was cut into rings about 0.475 centimeter (3/16 in.) wide for evaluation. The cuts were made with a dry hacksaw blade in the hot cell atmosphere using a niobium - 1 percent zirconium holding block to facilitate cutting the small ring samples. The saw cuts were filed and dry sanded to remove microcracks at the cut surfaces. The intent of the dry cutting and sanding was to keep moisture out of the T-111. The ring expansion test was conducted in an atmosphere of flowing dry argon. Strain was induced in the ring by forcing a hardened, tapered steel plug into a stainless-steel body (see fig. 9) at a constant rate. Figure 9 shows the ring and expander beyond the failure point of the ring. The radial strain rate in the ring was

was approximately 7.62×10^{-5} centimeter per second (3×10^{-5} in./sec).

Of the clad specimens that were tested, as-irradiated specimens from all but two pins failed in the range of 1 to 3 percent strain, indicating an embrittlement of the clad. The as-irradiated clad on the other two pins was ductile. One of these (a full length prototype pin) was irradiated for only 2000 hours. However, the other pin (a thin-clad, half-diameter pin) was irradiated for 13 000 hours (equivalent to 26 000 hr of burnup). No explanation can be given to account for the ductility of the clad on this pin while all other clads with significant burnup were embrittled. A similar pin run simultaneously in the same test hole was not tested for ductility, but a specimen from a reference-diameter pin in the same assembly failed at 3 percent strain.

Ring ductility specimens from most of the embrittled clads were given a vacuum heat treatment of 1 hour at 1900° F (1038° C). After this heat treatment, all but two of the clads were apparently as ductile as unirradiated control specimens. The embrittlement of the as-irradiated material is attributed to hydrogen from the n, p reaction (ref. 4) with the fuel and not to radiation damage. The vacuum heat treatment was effective in removing hydrogen. An additional specimen from one of the two clads that retained their embrittlement was given a vacuum heat treatment of 1 hour at 2550° F (1400° C). This heat treatment restored the ductility. These two pins were believed to have been contaminated with oxygen. An analysis by inert gas fusion on the 2550° F (1400° C) heat-treated clad showed nitrogen in the 38- to 40-ppm range and oxygen in the 81- to 103-ppm range. The effectiveness of the higher temperature heat treatment is attributed to redistribution of oxides concentrated in the grain boundaries. The available evidence indicates that the clad becomes brittle during the irradiation. To help confirm this, clad ductility tests were conducted on three unirradiated pins, one from each of the three phases of the test program, to determine whether the cladding is ductile as-received from the contractor. In all cases the clad was ductile.

The design of the test capsules is such that the hydrogen is retained inside the capsule by the cold capsule walls; it accumulates as it is generated; and it is gettered by by the T-111 cladding on cooling. In the actual reactor concept hydrogen would not accumulate but would permeate through the hot reactor pressure vessel and be lost to the surrounding vacuum of space. Hence, the hydrogen environment of the tests is not representative of the actual operating conditions. To avoid the nonrepresentative hydrogen environment and associated hydrogen embrittlement in test capsules, it is necessary to design them so that hydrogen is continuously removed at operating temperature. This might be achieved by the use of a hydrogen getter (e.g., zirconium at $\sim 600^{\circ}$ C (1100° F)) inside of the capsule.

The appearance of all the pins was bright and shiny when removed from the capsules, and, except for the two pins with axial cracks, the pins appeared to be the same as the

preirradiation pins. Figure 10 shows two of the four pins removed from capsule assemblies 012 and 013. These pins were irradiated for 7000 hours of real time.

Clad and Fuel Pellet Measurements

Fuel pin clad and fuel pellet dimensional measurements were made using an optical gage. The instrument could be read directly to 0.00254 millimeter (1×10^{-4} in.), and readings to 0.000254 millimeter (1×10^{-5} in.) could be estimated. Before either fuel pin clad or fuel pellet measurements, a calibrated standard was measured to verify the accuracy of the optical gage. Fuel-pin clad measurements were made at seven axial positions at both 0° and 90° (circumferentially) corresponding to the preirradiation measurement locations. Table VI lists the maximum and minimum values for the 0° and 90° positions for both the irradiated and preirradiated conditions.

As listed in the table none of the true time pins or pins accelerated by a factor of two (in burnup rate) showed any increase in clad diameter. The clad was not expected to increase in diameter because the fuel did not swell enough to fill the assembly clearance between fuel and clad as discussed in reference 4. However, measurements showed a decrease in clad diameter in some cases. As shown in the table the clad diameter decrease appeared to be as much as 0.0254 millimeter (1 mil). Part of the diameter decrease can be attributed to the final acid etch, which would account for approximately 0.00254 millimeter (0.1 mil). No measurements were made after the final cleaning and anneal. The reason for the remaining diametral decrease is not known.

Fuel pellet measurements were taken at the fuel pellet centerline, at both 0° and 90°, using the optical gage. The results of the fuel pellet measurements are given in table VII. The percent increase in fuel pellet diameters ranged from 0.10 to 2.22 percent. The fuel swelling was essentially isotropic, based on the length and diameter measurements.

Fuel swelling appeared to be primarily burnup dependent. Measurements of fuel pellets irradiated at real and accelerated times indicated that up to 2.0 percent burnup the net swelling was proportional to burnup as shown in figure 11. Note the high percent change in diameter $\Delta D/D$ value at 2.16 percent burnup where the clad had failed. This is pin 503I. In this pin the fuel swelling was significantly greater than in other pins.

The fuel pellets in most cases remained intact and were free from cracks. In an isolated case five of the six pellets from one pin developed axial cracks. This pin had been subjected to a severe overtemperature during operation. In no case did the fuel crumble or break into small pieces.

Fuel density measurements were made before and after irradiation on three fuel pins using a mercury pycnometer. The change in density is listed in table VII. On the

average, the fuel densities decreased by approximately 2 percent for burnups of up to 1 percent. Some additional density measurements were made using the immersion technique in carbon tetrachloride. These values are also listed in table VII. Curtailment of the program prevented additional measurements from being made using the mercury pycnometer.

Burnup Analysis

Cross sectional wafers were cut from selected positions of the fuel for destructive analysis. Some wafers were used to determine the average axial variation in burnup. Samples were taken from some of the wafers with a microdrill for determination of circumferential burnup. The primary method used to determine burnup of the fuel was the change in the uranium-236 to uranium-235 ratio ($^{236}U/^{235}U$) obtained from the mass spectrometer analysis. The maximum to minimum variations in the axial and circumferential burnup values were in most cases less than 1.25 and therefore are not tabulated. Listed in table IV are both the measured (mass spectrometer) and calculated burnup values. The calculated values were based on heat-transfer considerations. In general the measured values are slightly higher than the calculated values. The reason for this is primarily the method of calculation as discussed in reference 4. Some measured burnup values were not actually measured but were estimated, based on testing in the same test hole as a measured pin or based on the irradiation time and measured burnup in similar test holes.

A considerable amount of gamma scan work (unpublished) was performed in support of this program. Gamma scanning affords a rapid, nondestructive method of determining axial burnup variation in irradiated UN fuel pins. However, the data must be normalized to mass spectrometer burnup values in order to determine the absolute value of a gamma-scanned axial burnup profile. Typically, if several mass spectrometer determinations are performed on one fuel pin, the results should be applicable to other pins of the same composition. Gamma scanning along the fuel pin length for axial burnup determinations was conducted both with and without the T-111 clad. In the case without the cladding, the fuel pellets were removed from the clad, stacked on a rod as shown in figure 8, and inserted into a chuck positioner located within the gamma scan facility. For both axial scans (with or without clad), the fission product niobium-95 was used as the monitor. Figure 12 shows a typical comparison between axial burnup profiles determined by mass spectrometer and gamma scanning techniques. It is felt that gamma scanning provides an economical, reliable, and nondestructive method of obtaining additional burnup data from fuel pins.

The burnup value at a given axial location was determined within an error of 7 percent at the 2σ level when compared with the burnup value determined on the mass spectrometer.

Fission Gas Release

The amount of fission gas released from the fuel was obtained from measurements of krypton-85 activity in gas samples taken after puncture of the capsules or fuel pins, depending on whether the fuel pin leaked. A photograph of the capsule puncturing facility and the associated equipment to take gas samples and check for leaks is shown in figure 13.

Table VIII shows the results of a typical gas analysis taken from fuel pin 505E irradiated for 7000 hours (equivalent to 28 000 hr at the reference burnup rate). Also tabulated are the analyses for blank samples taken to check procedures and determine the extent of gaseous impurities that may be present due to sampling techniques. Two blank samples were taken, with and without the puncture drill running. No significant quantities of impurities were noted in the blank samples.

Difficulty was encountered in determining gas release from two fuel pins (503I and 503G) because of leaks. This resulted from cutting the thermocouple sheaths too close to the capsule. In these cases the fuel pin and tantalum-tipped thermocouple sheaths both leaked and allowed fission gas to diffuse through the tightly packed thermocouple insulation. Following this difficulty, the thermocouple leads were kept longer and cut just before puncture.

Measurements of krypton-85 activity in the gas samples were compared with values calculated from the measured average burnup and the yield of krypton-85 to obtain percent release from the fuel. Generally, the amount of fission gas released was small. With two exceptions, the amount released was less than 0.5 percent as shown in table IX. The two pins with amounts greater than 0.5 percent were both less than 2 percent. Even though the amount of fission gas release was small, there was a wide variation between pins irradiated under similar conditions. For example, the pin run to an equivalent burnup of 32 000 hours released seven times the amount of fission gas as the pin run for the equivalent of 35 400 hours. Both pins were run in the same test hole at about the same conditions. Also of the two pins run for an equivalent burnup of 28 000 hours, one pin released 10 times the amount of fission gas as the other. However, two pins irradiated for burnups equivalent to 28 400 hours each released approximately 0.1 percent of fission gas. The accuracy of the fission gas measurement was such that the relative difference between pins is real and not attributed to an error in measurement. The differences between pins may be due to the difference in the time of the occurrence of pin

leak (fission gas pressure decreasing when leaking into the capsule) or any internal cracks in the fuel.

Figure 14 shows a plot of percent fission gas release as a function of equivalent reference reactor irradiation hours (proportional to burnup). Below about 28 000 hours of equivalent reference burnup the amount of gas release is proportional to burnup. Beyond this point the interaction of the pressures exerted between fuel and clad probably alter the amount of fission gas release (the postirradiation examination revealed the fuel had swelled, closing the fuel clad gap of the two pins irradiated to 28 000 hours of equivalent burnup). Even using the increased amount of fission gas release and extrapolating to 50 000-hour life, the amount of fission gas release would be less than the 5 percent assumed for the reference design. The inset in figure 14 shows the ORR data point for 50 000-hour burnup (achieved in 10 000 hr) with less than a 5-percent fission gas release. The inset contains two Plum Brook data points beyond the range of the main plot.

Neutron Fluence

The total fluence incident on the fuel capsules is listed in table X. The values were either measured or calculated. The measured values were determined by counting flux wires attached to the capsule holder. The wire used for the thermal flux monitors was a 99-percent aluminum - 1-percent cobalt alloy. Stainless-steel and titanium were used as fast flux monitors. Data for capsules without measured values were obtained by using measured fluences in conjunction with unperturbed mockup reactor and Plum Brook Reactor measurements to correct for the various test hole locations.

Thermocouple Performance and Reliability

Thermocouple reliability is an important part of an irradiation program. It is felt that the thermocouple performance in this program was good. For example, 44 thermocouples were used on the capsules located in the reflector region of the reactor (see fig. 1), and only 1 thermocouple failed (11 426 hr of irradiation). In the lattice location (fig. 1) of the reactor there were 16 thermocouples of which 7 failed. (See table XI.) Reflector irradiation times reached 13 000 hours, and lattice irradiation times reached 8840 hours. No fluences were measured on the thermocouples; the maximum neutron fluences measured on the capsules are as follows:

Location	Maximum neutron fluences, nvt	
	Thermal	Fast (>0.1 MeV)
Reflector	4.3×10^{20}	3.0×10^{20}
Lattice	8.1×10^{20}	1.9×10^{20}

The higher incidence of thermocouple failure in the lattice location is not attributed to the higher neutron fluence. Rather several factors could have contributed to the higher failure rate for these thermocouples. Two of these factors are

(1) Handling of the experiment in the lattice location after each cycle for reactor fuel loading (this was not the case in the reflector region).

(2) Slightly higher operating temperature.

It is felt that the success of the thermocouples was due to the fabrication techniques used to form the ungrounded 5.08-centimeter (2-in.) long high-temperature zone and hot junction. Extreme care was taken to assure a loose fit between the wires, sheath, and ceramic both axially and radially (see fig. 15). Minimization of moisture content was also stressed. Detailed descriptions of the thermocouple assembly and fabrication procedures are given in appendix A of reference 4 and in reference 5. The two center thermocouples from capsule 503C irradiated for 8070 hours were recalibrated in a furnace within the hot lab. A temperature gradient closely simulating the inpile gradient was used. The thermocouples, when compared with the same type of unirradiated, unaged thermocouple, decalibrated (lower in temperature) a maximum of 27° C (38° F) at 1000° C (1832° F). The lead-to-sheath (insulation) resistance of the irradiated thermocouples was as high as or higher than the lead-to-sheath resistance of the unirradiated, unaged thermocouple. It is felt that the thermocouples suffered no significant irradiation damage and that decalibration was probably due to compositional changes in the thermoelements resulting from the temperature gradients.

Metallography

The amount of metallographic work conducted on the irradiated fuel pins was limited because of the termination of the program. Of the 27 fuel pins irradiated for varying lengths of time, only three were examined metallographically. These three were selected because of anomalies in the results of clad ductility tests from fuel pins 503C and 504B (see section Clad Ductility Tests) and to locate the leak in fuel pin 503B. These pins were irradiated simultaneously in the same test hole, and their irradiation history is given in tables III and IV.

The objective of the metallographic examination was to look for the following:

(1) Changes in structure of the fuel (e.g., grain growth, appearance of a second phase, fission gas bubbles; and void migration)

(2) Changes in structure of the cladding and liner (e.g., grain growth, abnormal precipitates, voids)

(3) Reaction areas in the fuel, liner, and/or cladding, possibly caused by chemical interactions between any two of the three under the conditions of the tests

(4) Other signs of fuel pin degradation such as cladding cracks, leak sources, and other material defects.

Three simulated fuel pins were also examined metallographically after aging out-of-pile for 2862 hours in helium filled capsules at 1038° C (1900° F). The metallography from the simulated pins served as a standard against which microstructural features of the irradiated fuel and cladding were compared.

Figure 16 shows the microstructure of the T-111 from the unirradiated thermally aged fuel pins. Some precipitates can be seen in the grain boundaries, which is typical of this material (ref. 10). No significant interstitial pickup occurred during this aging process (e.g., <30 ppm O_2, <20 ppm N_2).

The UN fuel structure from the simulated fuel pins shown in figure 17 at ×150 is typical of 94 percent theoretical density, unirradiated, thermally aged fuel. The fuel was viewed at ×500, and no second phases were visible. Also, there are no differences when compared to as-fabricated fuel.

The metallographic results on the three irradiated fuel pins are as follows:

Pin 503B. - This pin was subjected to metallographic examination with a large amount of work directed to the top end cap in search of a leak emanating from the thermocouple well. The generally clean appearance of the end cap is shown in figure 18. The area of the leak is shown in figure 19. This is a section of the wall of the machined thermocouple well that is near the final closure weld in the end cap. The large amount of grain pullout and separated grains could be due to either a void in the original material or from a contaminant penetrating the grain boundaries. The cause was not determined.

The fuel from this pin is shown in the unetched and etched condition in figure 20. The microstructure is not different from that of the unirradiated thermally aged fuel (fig. 17).

Cladding metallography indicated no reaction areas or grain growth. Also no unusual amounts of precipitates were seen in the cladding structure. Thus the cladding microstructure is essentially the same as that of thermally aged T-111.

Pin 503C. - The fuel from this pin is shown in figure 21 and appears unchanged from the thermally aged fuel; no fission gas bubbles or second phases are visible. Figure 22 shows the microstructure of the cladding from near the axial center of the pin.

Precipitates are more concentrated than those in figure 16, and the precipitates are seen in the grains and grain boundaries. Figure 23 shows the structure of the cladding taken near the top end of the fuel pin. It is evident that the amount of precipitate is greater in figure 22 than in figure 23. The results from the cladding ductility tests (see the section Clad Ductility Tests) and the inert gas fusion analysis for oxygen and nitrogen for this pin also indicate the possibility that the cladding was contaminated near the center of the fuel pin. The contaminate could be oxygen because the nitrogen concentration value is comparable to the value in the as-received material. The oxygen value increased from 30 to 40 ppm (by weight) in the as-received cladding material to 80 to 100 ppm in the irradiated cladding. Localized concentrations of oxygen within the ring ductility sample used for analysis may have been much higher. This could cause embrittlement as described in reference 11 if the cladding were contaminated while it was at a temperature near 1000° C (1832° F).

Pin 504B. - The fuel from this pin appears identical to the fuel from the previous two pins. The cladding from this fuel pin as shown in figure 24 indicates a heavier precipitate than found in either of the above pins. This precipitate has not been identified. It is believed that this also was contaminated by oxygen because the results of the ductility tests and oxygen and nitrogen analyses are similar to those from tests on the cladding of 503C. In addition, a 1-hour heat treatment of 1400° C (2550° F) restored the ductility as was described in reference 11.

The results of this optical metallographic examination are summarized as follows:

(1) No microstructural changes could be identified in any of the three fuel samples. No signs of grain growth, second phases, fission gas bubbles, and void migration could be found.

(2) No signs of grain growth or voids was noticed in the cladding material. Some precipitates were observed in two of the fuel pins but this will be covered in item (4).

(3) There was no evidence of reactions which would indicate incompatibility between the components of the fuel pins.

(4) Results of ductility tests, inert fusion gas analyses, and metallography (heavy precipitates in pins 503C and 504B) taken together suggest that a small amount of oxygen contamination of the T-111 cladding occurred. Of the 27 irradiated pins these fuel pins (503C and 504B) are the only ones that appeared to have contaminated cladding. The third fuel pin (503B) exhibited a microstructure similar to that of thermally aged unirradiated T-111 cladding.

CONCLUDING REMARKS

This report described the examination of 27 fuel pins irradiated at real and accelerated burnup rates for times ranging to 13 000 hours at 990° C (1815° F) clad tempera-

ture. Although there was no evidence of incompatibilities between fuel, liner, and clad, the clad was embrittled during irradiation resulting in fission gas leaks in about half of the pins. More specific observations follow:

1. Most of the fuel pin clad was embrittled during irradiation. The embrittlement is attributed to hydrogen from the n, p reaction with the fuel.

2. Fission gas leaks were noted in 13 of 22 fuel pins tested for leaks. Two of thirteen had visible cracks in the clad. Leaks in other fuel pins are thought to be in the end caps.

3. No incompatibilities between fuel, liner, and clad were in evidence.

4. Fuel swelling was essentially isotropic, based on length and diameter measurements. Percent change in diameter ($\Delta D/D$) of the fuel was approximately proportional to burnup. The fuel density of the few pellets measured showed a decrease of 2 percent for 1 percent burnup.

5. The fuel pellets in most cases remained intact and free from cracks.

6. The fuel burnup as determined by mass spectrometer measurements was in good agreement with values obtained from heat transfer calculations.

7. Standard type K thermocouples modified in the hot zone were used for this test program. The reflector thermocouples (1 out of 44 failed) were more reliable than the lattice thermocouples (7 out of 16 failed). We feel that two of the factors contributing to the higher failure rate of the lattice thermocouples were experiment handling and higher operating temperature rather than the higher neutron fluence.

8. The amount of fission gas released from the fuel was low, generally less than 0.5 percent.

The program was terminated before any of the fuel pins achieved the reference design burnup of 2.73 atom percent. Nevertheless, under the conditions of the tests, the fuel appears promising for space reactor application because the fuel was free from cracks and had a low amount of fission gas release. Unfortunately, the design of the capsules was such that the generated hydrogen was retained inside the capsules by the cold capsule walls leading to embrittlement of the T-111. In the actual reactor concept hydrogen would not accumulate but permeate through the hot reactor pressure vessel. Therefore, the hydrogen environment is not representative of the actual operating conditions. Even though the clad was embrittled during irradiation, it did accommodate some strain before failing in the ductility tests.

Further, the neutron spectrum was not representative of the fast damage that the capsules would experience under reference design conditions. At the termination of the

program, fast damage test pins and associated thermal neutron shields were being designed and fabricated.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, September 10, 1973,
503-25.

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TABLE I. - TEST PROGRAM VARIABLES (NOMINAL VALUES)

Clad material	T-111
Fuel:	
Material (at 94 % of theoretical density[a])	Uranium nitride
Form	Cored pellets
Clad temperature[b], ^{o}C (^{o}F)	990±40 (1815±72)
Fuel-pin diameter with regular clad, cm (in.):	
Reference-diameter pin	1.90 (0.750)
Half-diameter pin	0.95 (0.375)
Fuel-pin diameter with thin clad, cm (in.):	
Reference-diameter pin	1.81 (0.714)
Half-diameter pin	0.905 (0.357)
Fuel pellet diameter, cm (in.):	
Reference-diameter pin	1.58 (0.622)
Half-diameter pin	0.787 (0.310)
Fuel pellet inside diameter, cm (in.):	
Reference-diameter pin	0.510 (0.201)
Half-diameter pin	0.280 (0.110)
Clad thickness for reference-diameter pin, cm (in.):	
Regular	0.147 (0.058)
Thin	0.102 (0.040)
Clad thickness for half-diameter pin, cm (in.):	
Regular	0.074 (0.029)
Thin	0.051 (0.020)
Tungsten compatibility liner thickness, cm (in.):	
Reference-diameter pin	0.0127 (0.005)
Half-diameter pin	0.0076 (0.003)
Burnup rate, fissions/cm^3 sec:	
Reference (true time)	[c]0.5×10^{13}
Acceleration factor 2	1.0×10^{13}
Acceleration factor 4	2.0×10^{13}

[a]Theoretical density, 14.32 g/cm^3.

[b]Full length pin temperature variation, ±80^{o} C (±144^{o} F).

[c]Corresponds to 141 W/cm^3 of fuel using 176 MeV per fission for the PBRF.

TABLE II. - PLUM BROOK IRRADIATIONS

[All pins have tungsten liners, 0.127 mm (5 mil) thick for reference-diameter pins and 0.076 mm (3 mil) thick for half-diameter pins. Nominal cold diametral clearance between clad and fuel, 0.114 mm (4.5 mil) for reference-diameter pins and 0.089 mm (3.5 mil) for half-diameter pins. Diametral clearance between fuel and clad at temperature, 0.0636 mm (2.5 mil) for both size pins.]

Capsule identification	Fuel pin number	Nominal pin diameter		Actual clad thickness		Minimum planned irradiation time, hr	Relative burnup rate[a]	Fuel length		Clad temperature		Actual irradiation time, hr
		cm	in.	cm	in.			cm	in.	^oC	^oF	
001	504B	1.81	0.714	0.102	0.040	5 000	1	5.72	2.25	990	1815	8 070
	503B	.905	.357	.051	.020	5 000	2					8 070
	503C	.905	.357	.051	.020	5 000	2					8 070
004	502B	1.90	.750	.147	.058	10 000	1					883
	501D	.95	.375	.069	.027	15 000	2					12 090
	501F	.95	.375	.069	.027	15 000	2					12 090
005	502C	1.90	.750	.147	.058	20 000	1					9 386
	501C	.95	.375	.069	.027	20 000	2					11 630
	501E	.95	.375	.069	.027	20 000	2					11 630
003	504E	1.81	.714	.102	.040	15 000	1					13 000
	503D	.905	.357	.051	.020		2					13 000
	503E	.905	.357	.051	.020		2					13 000
006	504D	1.81	.714	.102	.040		1					6 650
	503F	.905	.357	.051	.020		2					6 650
	503H	.905	.357	.051	.020		2					6 650
002	502A	1.90	.750	.147	.058	20 000	1					6 925
	501A	.95	.375	.069	.027	20 000	2					6 925
	501B	.95	.375	.069	.027	20 000	2					6 925
010	503I	.905	.357	.051	.020	10 000	4					8 163
	503G					10 000						8 840
011	505E					12 500						7 000
	505F					12 500						7 000
012	507C	.95	.375	.069	.027	13 000						7 323
	507D	.95	.375	.069	.027	13 000						7 323
013	509A	.905	.357	.051	.020	12 500						7 100
	509B	.905	.357	.051	.020	12 500						7 100
020	510A	1.90	.750	.147	.058	20 000	1	38.1	15.0			1 990
030	505A	.905	.357	.069	.027	14 300	3.5	5.72	2.25	870	1600	1 150
	505B	.905	.357	.069	.027	14 300	3.5	5.72	2.25	870	1600	1 150

[a]The nominal burnup rate of 1 corresponds to 0.5×10^{13} fissions/cm^3 sec. Fission power density is based on 176 MeV/fission in the PBRF. Therefore, a relative burnup rate of 1 corresponds to 141 W/cm^3 of fuel.

TABLE III. - SUMMARY OF FUEL PIN TEMPERATURES AND THERMAL CYCLES

Fuel pin number	Average centerline temperature, K			Total hours irradiation	Number of thermal cycles	Hours of irradiation at average fuel-pin centerline temperatures between -			
	Top thermo-couple	Bottom thermo-couple	Fuel pin average			1350 to 1425 K	1250 to 1350 K	1150 to 1250 K	<1150 K
504B	1364	1370	1367	8 070	82	6579	1 491	0	0
503B	1271	1242	1257	8 070	82	108	4 201	3683	78
503C	1260	1247	1254	8 070	82	331	3 435	4178	126
502B	1390	1316	1354	883	14	431	452	0	0
501D	1326	1332	1329	12 100	76	2	11 966	107	25
501F	1334	1313	1323	12 100	76	0	11 929	64	107
502C	1345	1379	1362	19 386	66	7817	1 492	47	27
501C	1245	1302	1274	11 623	74	0	4 415	4099	3109
501E	1272	1266	1269	11 623	74	0	3 082	7030	1511
504E	1340	1325	1332	13 000	82	6804	6 108	88	0
503D	1311	1316	1313	13 000	82	22	12 786	192	0
503E	1288	1307	1297	13 000	82	50	11 003	1903	44
504D	1345	1342	1343	6 650	34	5486	1 067	94	3
503F	1306	1256	1281	6 650	34	3598	651	2397	4
503H	1214	1237	1225	6 650	34	652	1 921	3211	866
502A	1337	1310	1322	6 930	33	1660	4 870	400	0
501A	1353	1344	1348	6 930	33	3300	3 610	20	0
501B	1352	1336	1344	6 930	33	3620	3 310	0	0
503G	1332	1362	1347	8 750	70	3491	4 313	691	255
503I	1291	1355	1323	8 163	66	1593	4 692	1759	120
505E	1343	1225	1284	7 091	50	3048	3 382	629	32
505F	1365	1230	1298	7 091	50	4662	1 942	479	8
507C	1337	1348	1342	7 323	49	1747	5 574	2	0
507D	1331	1345	1338	7 323	49	3130	4 193	----	----
509A	1293	1365	1329	7 065	50	2206	3 546	1207	107
509B	1319	1366	1342	7 065	50	2193	4 406	466	0
505A	1238	1266	1252	1 147	--	----	------	----	----
505B	1243	1275	1259	1 147	--	----	------	----	----
510A	1285	1350	1318	1 990	--	----	------	----	----

TABLE IV. - POSTIRRADIATION EXAMINATION OF FUEL PINS IRRADIATED IN PLUM BROOK REACTOR

[Fuel, uranium nitride (94 percent of theoretical density); clad material, T-111; clad temperature, 990° C (1815° F); regular clad thickness, 0.14 cm (58 mil) for reference-diameter pins and 0.069 cm (27 mil) for half diameter pins; clad thickness, 0.102 cm (40 mil) for reference-diameter pins and 0.051 (20 mil) for half-diameter pins.]

Capsule identi-fica-tion	Fuel pin number	Pin diameter	Clad thickness		Time, hr		Burnup[a], at. %		Pin integrity	Fission gas release, percent	Fuel swelling, percent		Clad ring ductility, percent ΔD/D to failure	
			cm	in.	Real	Accelerated[b]	Calculated	Measured			ΔD/D	ΔL/L	As irra-diated	After heat treatment
020	510A	Reference	0.147	0.058	1 990	1 990	0.11	-----	No leak	----	-----	----	>5	----
030	505A	Half	.051	.020	1 150	4 000	.22	-----	Leak, LU[c]	0.15	-----	----	-----	----
	505B	Half	.051	.020	1 150	4 000	.21	-----	-------------	----	-----	----	-----	----
006	504D	Reference	.102	.040	6 650	6 650	.38	-----	-------------	----	-----	----	-----	----
002	502A		.147	.058	6 930	6 930	.36	[d]0.40	Leak, LU	.03	-----	----	-----	----
001	504B		.102	.040	8 070	8 070	.47	.47	No leak	.02	[e]0.1	0.27	1.5	[f]3
													3.0	>[g]5
003	504E		.102	.040	13 000	13 000	.73	[d].77	No leak	----	-----	----	3.0	----
006	503H	Half	.051	.020	6 650	13 300	.61	-----	-------------	----	-----	----	-----	----
	503F		.051	.020	6 650	13 300	.66	-----	-------------	----	-----	----	-----	----
002	501B		.069	.027	6 930	13 850	.75	-----	-------------	----	-----	----	-----	----
	501A		.069	.027	6 930	13 850	.76	-----	No leak	----	-----	----	-----	----
001	503C		.051	.020	8 070	16 140	.78	.92	No leak	.02	.39	.45	3	>[f]4
	503B		.051	.020	8 070	16 140	.78	.89	Leak, thermo-couple well	.05	.45	.53	0	>[f]5
005	501E		.069	.027	11 630	23 260	1.10	-----	No leak	----	-----	----	-----	----
	501C				11 630	23 260	1.10	-----	No leak	----	-----	----	-----	----
004	501F				12 090	24 180	1.29	[d]1.36	Leak, LU	.10	-----	----	-----	----
	501D				12 090	24 180	1.30	-----	No leak	----	-----	----	-----	----
003	503E		.051	.020	13 000	26 000	1.34	[d]1.46	Leak, LU	.09	1.0	----	>5	----
	503D				13 000	26 000	1.36	[d]1.46	Leak, LU	.09	-----	----	-----	----
011	505E				7 000	28 000	1.47	1.85	Leak, end cap	.22	.97	----	2	>[f]5
	505F				7 000	28 000	1.49	1.85	Leak, LU	1.91	-----	----	-----	----
013	509A				7 100	28 400	1.43	[d]1.88	Leak, LU	.09	-----	----	-----	----
	509B				7 100	28 400	1.46	[d]1.88	Leak, LU	.13	.7	----	-----	----
012	507C		.069	.026	7 300	29 200	1.41	-----	No leak	----	-----	----	-----	----
	507D				7 300	29 200	1.47	[d]1.94	Leak, LU	.22	1.0	----	-----	----
010	503I		.051	.020	8 160	32 640	1.68	[d]2.16	Clad crack	1.29	2.2	1.8	-----	----
	503G		.051	.020	8 840	35 360	1.87	[d]2.34	Clad Crack	.18	-----	----	-----	----

[a]Reference reactor burnup, 2.73 at. % in 50 000 hr.
[b]Accelerated time calculated from real time using nominal acceleration factors.
[c]Location unknown.
[d]Burnup value time-ratioed from selective measured pins.
[e]One pellet.
[f]Heat treatment, 1 hr at 1038° C (1900° F).
[g]Heat treatment, 1 hr at 1400° C (2550° F).

TABLE V. - RADIAL TEMPERATURE DROP ACROSS FUEL PIN FOR BEGINNING OF LIFE CONDITIONS

Pin diameter	Burnup rate	Fuel	Pellet-clad clearance gap	Clad	Total
		Temperature drop, T, °C (°F)			
Reference	Reference	70	44	14	128 (230)
Half	Acceleration factor 2	32	40	6	78 (141)
Half	Acceleration factor 4	63	80	12	155 (280)

TABLE VI. - CLAD DIAMETER MEASUREMENTS

(a) SI units

Fuel pin number	Postirradiation diameters, cm					Preirradiation diameters, cm					Change in diameter, ΔD, cm	Percent change in diameter, ΔD/D
	0^{o}		90^{o}		Average	0^{o}		90^{o}		Average		
	Max.	Min.	Max.	Min.		Max.	Min.	Max.	Min.			
503B	0.9154	0.9139	0.9157	0.9144	0.9147	0.9162	0.9152	0.9160	0.9149	0.9154	-0.00076	-0.08
503C	.9152	.9129	.9126	.9134	.9134	.9167	.9154	.9154	.9162	.9160	-.00254	-.28
504B	1.8202	1.8189	1.8181	1.8192	1.8192	1.8224	1.8217	1.8219	1.8212	1.8219	-.00279	-.15
503E	.9154	.9144	.9147	.9136	.9144	.9157	.9154	.9157	.9152	.9154	-.001016	-.11
504E	1.8212	1.8209	1.8085	1.8209	1.8212	1.8219	1.8230	1.8224	1.8219	1.8224	-.00127	-.07
501F	.9479	.9470	.9474	.9470	.9474	.9484	.9474	.9482	.9477	.9472	-.00025	-.03
501E	.9487	.9482	.9482	.9479	.9482	.9487	.9485	.9492	.9485	.9485	-.00025	-.03
503I	.9286	.9230	.9302	.9271	.9274	.9154	.9149	.9154	.9149	.9149	.01245	1.36
503G	.9208	.9149	.9182	.9152	.9175	.9157	.9149	.9157	.9152	.9154	.00203	.20
505E	.9164	.9154	.9175	.9149	.9159	.9159	.9149	.9154	.9147	.9152	.00076	.08
505F	.9157	.9154	.9157	.9149	.9157	.9162	.9149	.9154	.9152	.9152	.00051	.055
509A	.9150	.9139	.9154	.9152	.9149	.9152	.9149	.9149	.9147	.9149	0	0

(b) U.S. customary units

Fuel pin number	Postirrandiation diameters, in.					Preirradiation diameters, in.					Change in diameter, ΔD, mils	Percent change in diameter, ΔD/D
	0^{o}		90^{o}		Average	0^{o}		90^{o}		Average		
	Max.	Min.	Max.	Min.		Max.	Min.	Max.	Min.			
503B	0.3604	0.3598	0.3605	0.3600	0.3601	0.3607	0.3603	0.3606	0.3602	0.3604	-0.3	-0.08
503C	.3603	.3594	.3593	.3596	.3596	.3609	.3604	.3604	.3607	.3606	-1.0	-.28
504B	.7166	.7161	.7163	.7158	.7162	.7175	.7172	.7173	.7170	.7173	-1.1	-.15
503E	.3604	.3600	.3601	.3597	.3600	.3605	.3604	.3605	.3603	.3604	-.4	-.11
504E	.7170	.7169	.7120	.7169	.7170	.7173	.7177	.7175	.7173	.7175	-.5	-.07
501F	.3732	.3728	.3730	.3728	.3730	.3734	.3730	.3733	.3729	.3731	-.1	-.03
501E	.3735	.3733	.3733	.3732	.3733	.3735	.3734	.3737	.3734	.3734	-.1	-.03
503I	.3656	.3634	.3662	.3650	.3651	.3604	.3602	.3604	.3602	.3602	4.9	1.36
503G	.3625	.3602	.3615	.3603	.3612	.3605		.3605	.3603	.3604	.8	.20
505E	.3608	.3604	.3612	.3602	.3606	.3606		.3604	.3601	.3603	.3	.08
505F	.3605	.3604	.3605	.3602	.3605	.3607		.3604	.3603	.3603	.2	.055
509A	.3602	.3598	.3604	.3603	.3602	.3603		.3602	.3601	.3602	0	0

TABLE VII. - FUEL PELLET MEASUREMENTS

(a) SI units

Fuel pin number	Postirradiation diameters, cm			Preirradiation diameters, cm			Average change in diameter, ΔD, cm	Average percent change in diameter, ΔD/D	Change in length, ΔL, cm	Percent change in length, ΔL/L	Average percent change in density, Δρ/ρ	Number of pellets measured
	Max.	Min.	Average	Max.	Min.	Average						
503B	0.7930	0.7887	0.7905	0.7887	0.7851	0.7869	0.0035	0.445	0.0051	0.525	2.1	6
503C	.7905	.7861	.7899	.7874	.7861	.7869	.0030	.387	.0043	.456	1.9	6
504B[a]	1.5827	1.5812	1.5812	1.5812	1.5812	1.5812	.0015	.10	.0025	.300	2.0	1
503E	.7968	.7955	.7963	.7892	.7879	.7882	.0081	1.0	------	-----	[b]5.3	1
502C	1.5878	1.5847	1.5865	1.5799	1.5786	1.5794	.0071	.45	.0071	.50	---	2
503I	.8085	.8024	.8052	.7892	.7861	.7877	.0175	2.22	.0172	1.81	---	6
505E	.7976	.7943	.7955	.7879	.7874	.7879	.0076	.97	------	----	---	6
507D	.7960	.7955	.7958	.7879	.7879	.7879	.0078	1.0	------	----	[b]4.1	1
509B	.7973	.7915	.7935	.7877	.7877	.7877	.0058	.74	------	----	[b]2.1	1

(b) U.S. customary units

Fuel pin number	Postirradiation diameters, in.			Preirradiation diameters, in.			Average change in diameter, ΔD, mil	Average percent change in diameter, ΔD/D	Change in length, ΔL, mil	Percent change in length, ΔL/L	Average percent change in density, Δρ/ρ	Number of pellets measured
	Max.	Min.	Average	Max.	Min.	Average						
503B	0.3122	0.3105	0.3112	0.3105	0.3091	0.3098	1.4	0.445	2	0.525	2.1	6
503C	.3112	.3095	.3110	.3100	.3095	.3098	1.2	.387	1.7	.456	1.9	6
504B[a]	.6231	.6225	.6225	.6225	.6225	.6225	.6	.10	1.0	.300	2.0	1
503E	.3137	.3132	.3135	.3107	.3102	.3103	3.2	1.0	---	-----	[b]5.3	1
502C	.6251	.6239	.6246	.6220	.6215	.6218	2.8	.45	2.8	.50	----	2
503I	.3183	.3159	.3170	.3107	.3095	.3101	6.9	2.22	6.8	1.81	----	6
505E	.3140	.3127	.3132	.3102	.3100	.3102	3.0	.97	---	-----	----	6
507D	.3134	.3132	.3133	.3102	.3102	.3102	3.1	1.0	---	-----	[b]4.1	1
509B	.3139	.3116	.3124	.3101	.3101	.3101	2.3	.74	---	-----	[b]2.1	1

[a]Only one pellet; others cracked.
[b]Carbon tetrachloride method.

TABLE VIII. - GAS COMPOSITION

OF CAPSULE PUNCTURE

Gas	Fuel pin 505E	Blank without drill running	Blank with drill running
	Content, vol. %		
Hydrogen	0.29	0.02	0.03
Helium	98.1	98.0	99.6
Methane	.09	.01	.01
Water	.1	.1	.2
Oxygen	.06	.39	.06
Nitrogen	.47	1.55	.10
Argon	.01	.02	.01
Carbon dioxide	.01	.01	.01
Krypton	.14	[a]ND	ND
Xenon	.85	ND	ND

[a]Not detected.

TABLE IX. - FISSION GAS RELEASE MEASUREMENTS

Fuel pin number	Fissions per cm^3 (a)	Fission gas (^{85}Kr) activity, dis/min	Fission gas released, percent (b)	Fuel pin number	Fissions per cm^3 (a)	Fission gas (^{85}Kr) activity, dis/min	Fission gas released, percent (b)
503C	2.57×10^{20}	0.395×10^{8}	0.02	501B	2.48×10^{2}	---------	-----
503B	2.59	1.10	.05	501A	2.51	---------	-----
504B	1.54	.748	.02	502A	1.25	1.33×10^{8}	0.03
503E	4.38	3.44	.09	503I	5.57	76.10	1.29
503D	4.49	3.44	.09	503G	6.19	11.20	.18
504E	2.41	2.75	----	505E	4.85	11.00	.22
501F	4.24	3.74	.10	505F	4.93	96.00	1.91
501D	4.28	--------	----	507C	4.66	---------	-----
501E	3.63	--------	----	507D	4.87	12.00	.22
501C	3.65	--------	----	509A	4.74	4.51	.088
503H	2.02	--------	----	509B	4.81	6.55	.13
503F	2.18	--------	----	505A	.71	.125	.15
504D	1.24	--------	----	505B	.70	---------	-----

[a]Calculated based on average fuel center temperature and heat-transfer considerations.
[b]Based on ^{85}Kr activity.

TABLE X. - NEUTRON FLUENCE CALCULATION RESULTS

Capsule identification	Fuel pin number	Irradiation time, hr	Average fast fluence ($E > 0.1$ MeV)		Average thermal fluence	
			nvt	Error, ± %	nvt	Error, ± %
001	503C	8 070	[a]1.4×10^{20}	25	[a]3.1×10^{20}	10
	503B	8 070	[a]1.5	25	[a]3.1	10
	504B	8 070	[a]1.5	25	[a]2.7	10
003	503E	13 000	[a]2.3	25	[a]4.3	10
	503D	13 000	2.3	25	4.3	10
	504E	13 000	2.3	25	3.7	10
004	501F	12 092	3.0	35	4.0	15
	501D	12 092	3.0	35	4.0	15
	502B	880	--------	---	----	---
005	501E	11 630	2.9	25	3.9	10
	501C	11 630	2.9	25	3.9	10
	502C	9 389	[a]2.3	25	[a]2.9	10
006	503H	6 650	0.9	35	2.4	15
	503F	6 650	.9	35	2.4	15
	504D	6 650	.9	35	2.1	15
002	501B	6 925	1.2	35	2.5	15
	501A	6 925	1.2	35	2.5	15
	502A	6 925	1.2	35	2.2	15
010	503G	8 840	[a]18.9	25	[a]8.1	10
	503I	8 163	[a]14.4	25	[a]7.6	10
011	505E	6 993	[a]11.4	25	[a]6.4	10
	505F	6 993	[a]9.9	25	[a]7.2	10
012	507C	7 323	14.3	35	7.0	15
	507D	7 323	14.3	35	7.0	15
013	509A	7 097	10.8	35	6.7	15
	509B	7 097	10.8	35	6.7	15
030	505A	1 150	1.7	35	1.1	15
	505B	1 150	1.7	35	1.1	15
020	510A	1 990	0.5	40	0.5	25

[a]Measured value.

TABLE XI. - TIME TO THERMOCOUPLE FAILURE

Fuel pin number	Location	Time to failure, hr
503I	Bottom	3 344
	Top	8 158
505E	Bottom	2 174
505F	Bottom	5 015
507D	Top	3 900
507C	Top	6 583
509A	Bottom	2 108
503D[a]	Top	11 426

[a]Thermocouple on capsule in reflector region; all others located in lattice region.

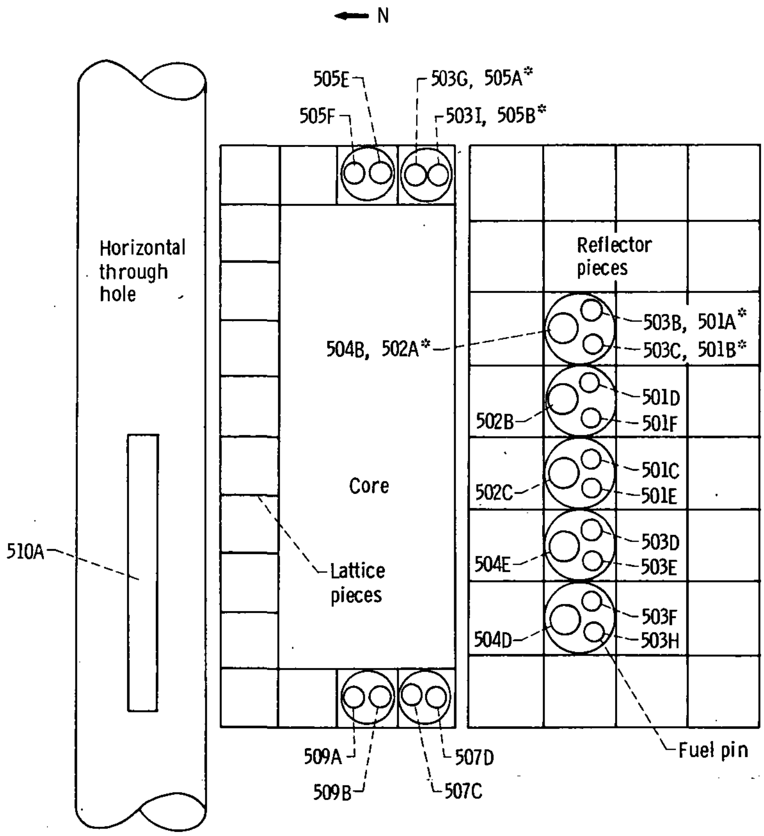
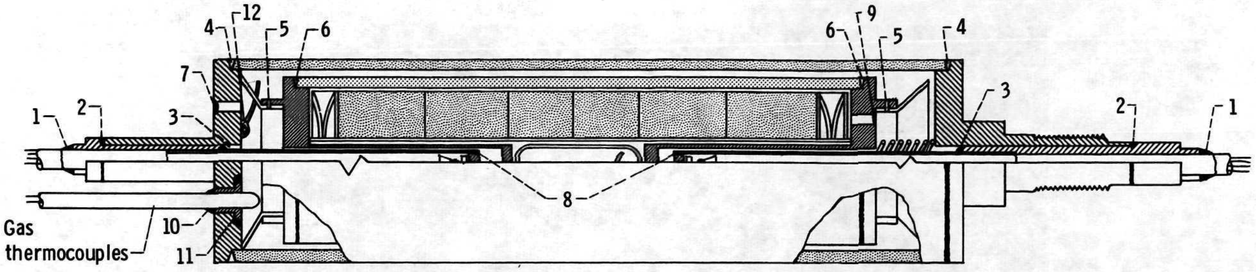


Figure 1. - Schematic horizontal plan view of Plum Brook Reactor showing fuel pin test locations. Asterisks denote two capsule assemblies irradiated in same test hole (not simultaneously).



1 Thermocouple sheath to thermocouple well (EB weld)
2 Thermocouple well to capsule end cap (EB weld)
3 Thermocouple well, stainless steel to tantalum transition (EB weld)
4 Capsule end cap to capsule tube (EB weld)
5 Fuel pin spacer to fuel pin (EB tack weld)
6 Fuel pin end cap to fuel pin tube (EB weld)
7 Capsule final closure (GTA weld)
8 Thermocouple well, tantalum plug to tantalum tubing (EB weld)
9 Fuel pin final closure (GTA weld)
*10 Gas thermocouple sheath to adaptor (EB weld)
*11 Gas thermocouple adaptor to capsule end cap (EB weld)
12 Positioning wire to capsule end cap (GTA weld)

Figure 2. - Weld schematic for capsule assemblies. (Asterisks denote reference-diameter capsule only.)

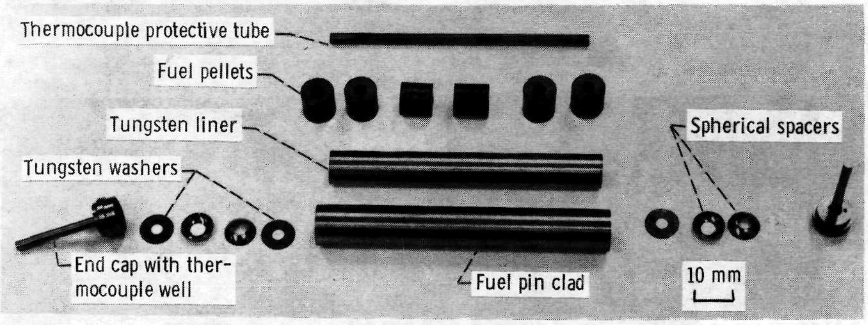


Figure 3. - Half-diameter size fuel pin components.

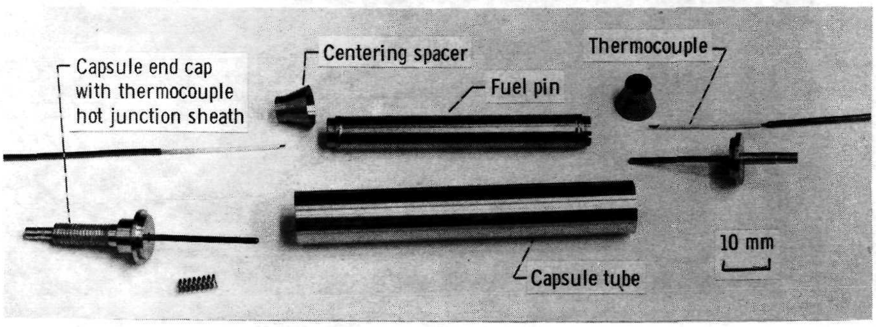


Figure 4. - Half-diameter size fuel pin, capsule components, and instrumentation.

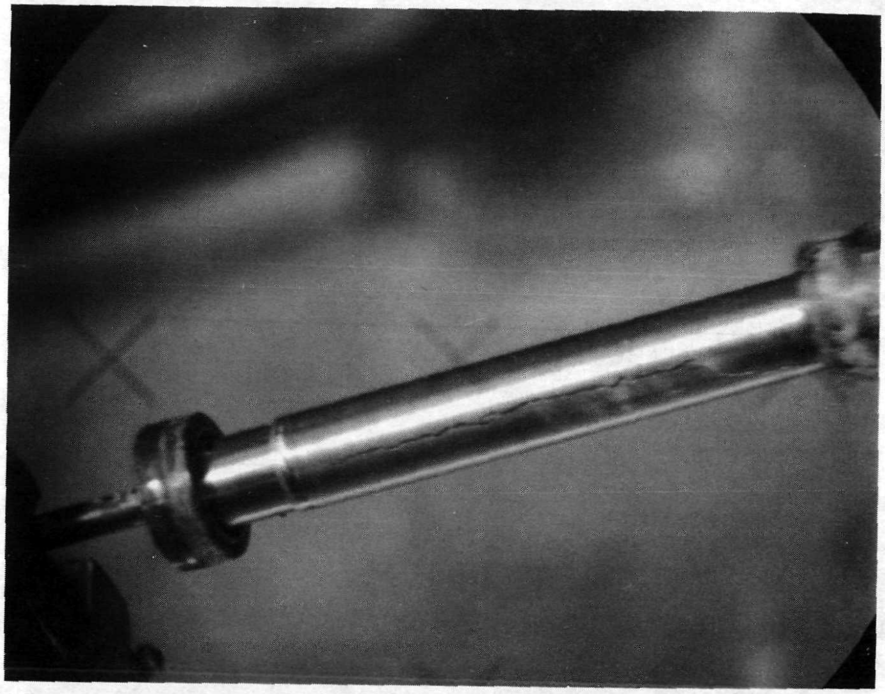


Figure 5. - Axial clad crack on fuel pin 503-I.

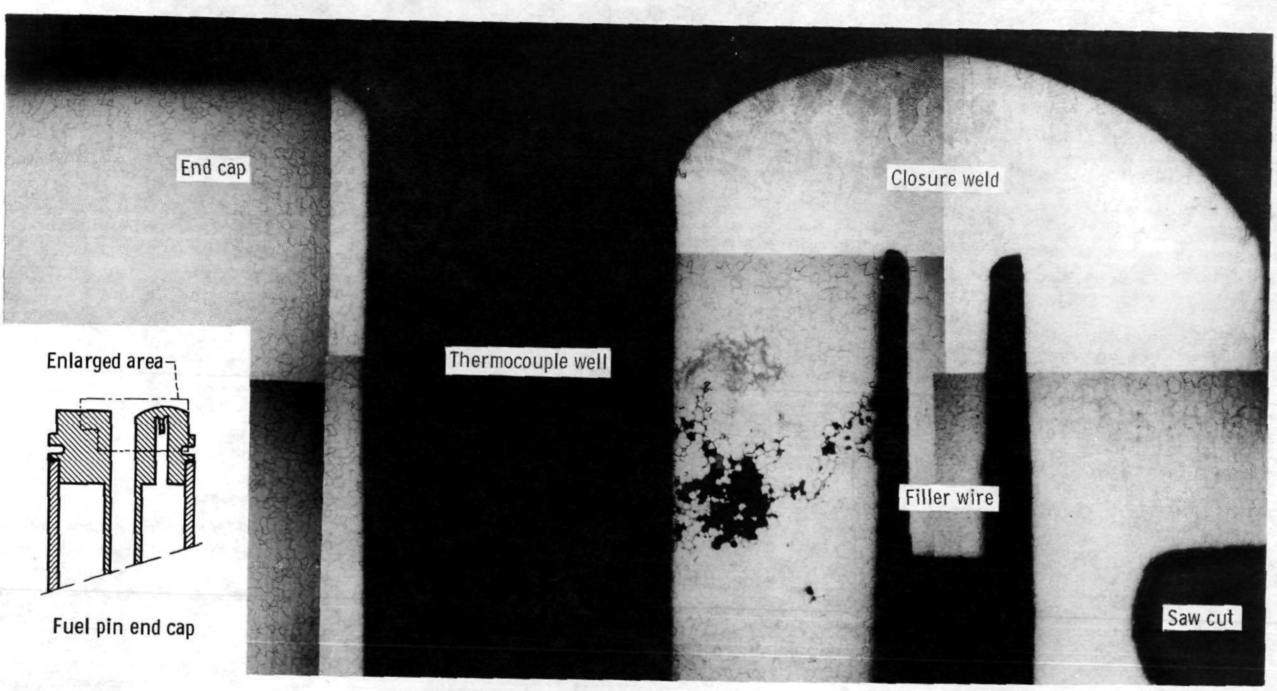


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Figure 6. - Photomicrograph of fuel pin end cap showing leak path between thermocouple well and closure hole. X40.

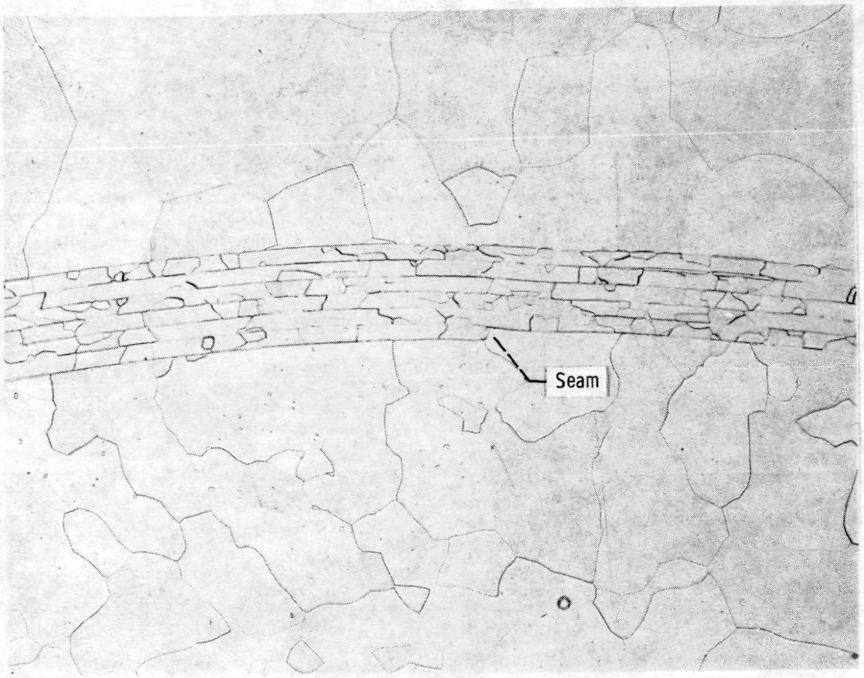


Figure 7. - Photomicrograph of typical tungsten liner made from 0.025 millimeter (1-mil) thick tungsten foil.

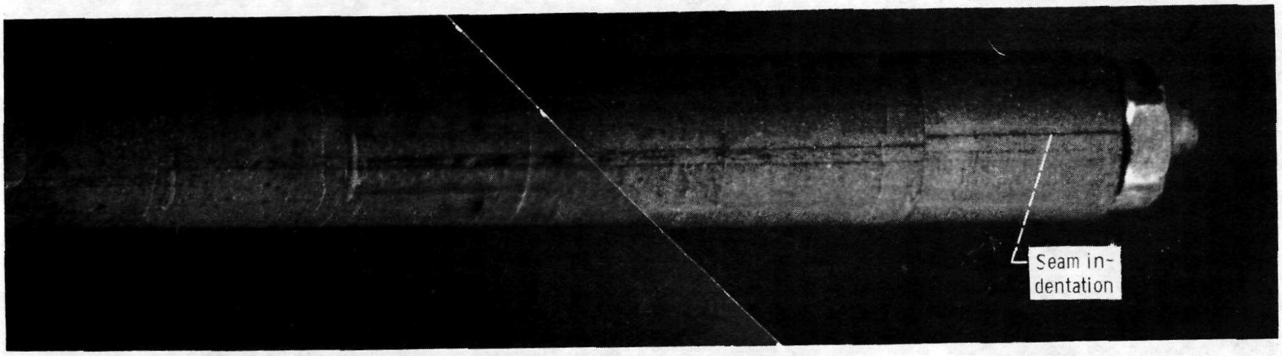


Figure 8. - Fuel stack from pin 505-E showing fuel indentation caused by seam of tungsten liner.

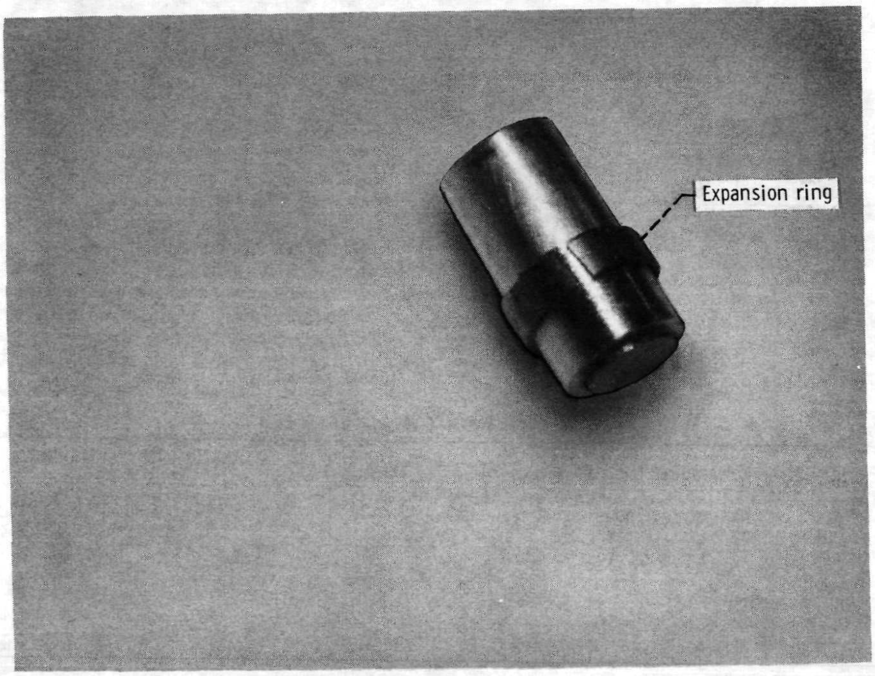


Figure 9. - Hardware for T-111 expansion ring ductility test showing typical brittle fracture.

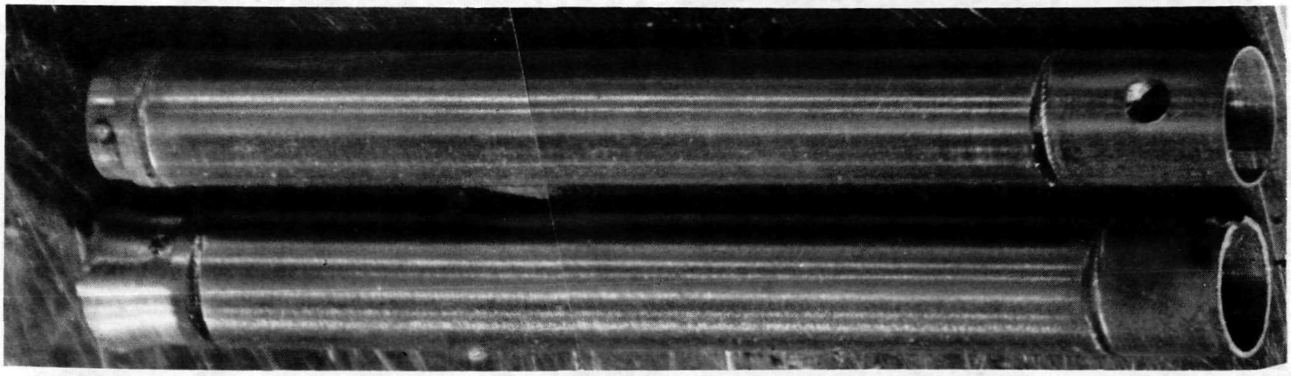


Figure 10. - Typical appearance of fuel pins irradiated for 7000 hours.

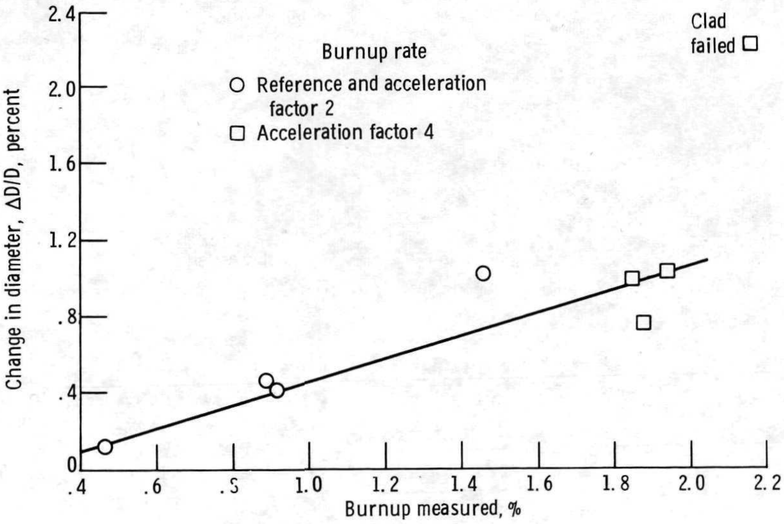


Figure 11. - Fuel percent change in diameter as function of fuel burnup.

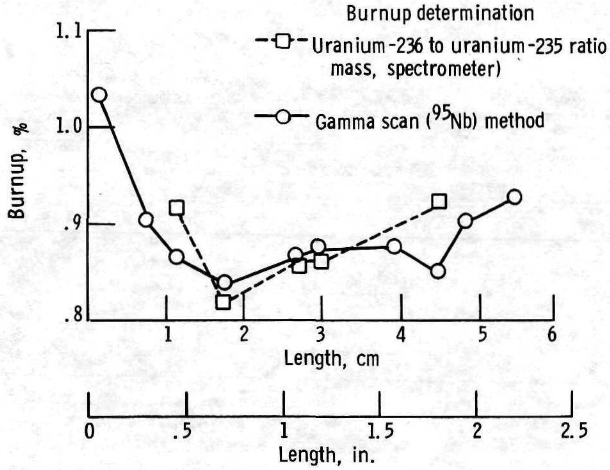


Figure 12. - Burnup determination comparison for fuel pin 503B.

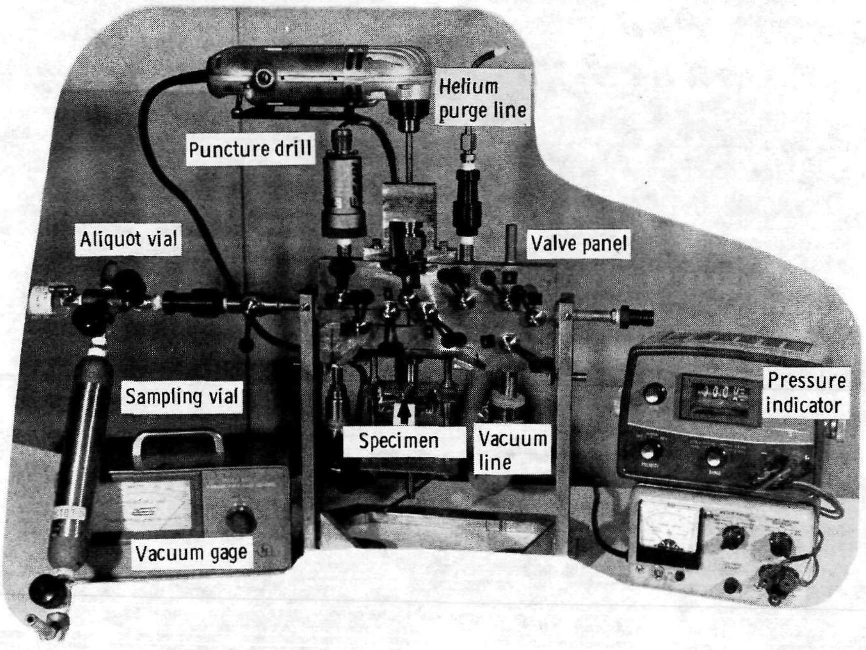


Figure 13. - Puncture rig.

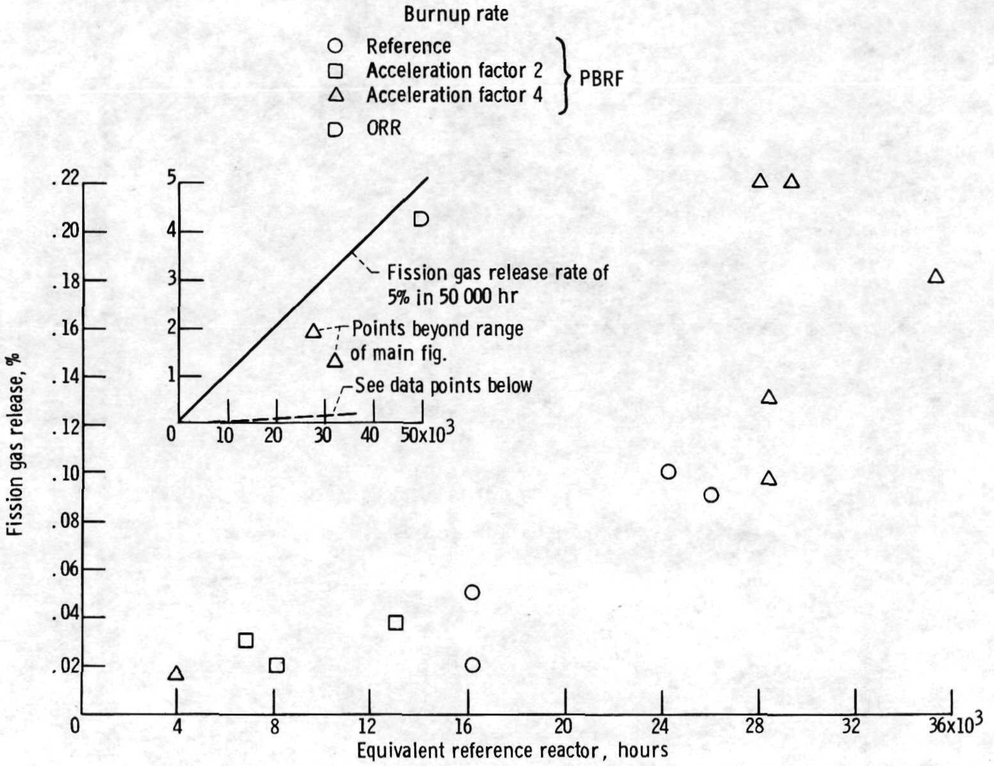


Figure 14. - Fission gas release for equivalent reference reactor hours.

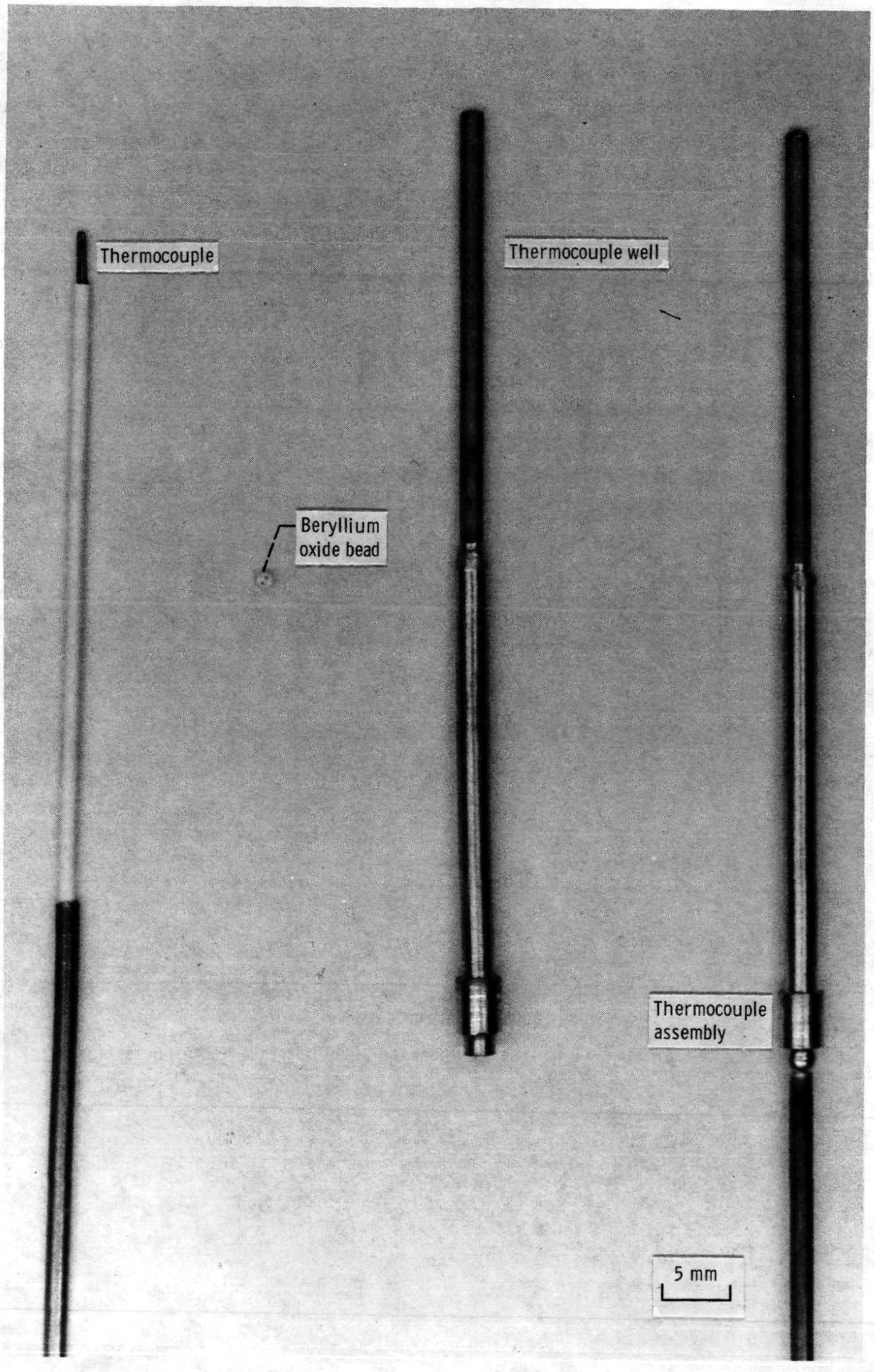


Figure 15. - Thermocouple subassembly showing hot junction fabrication technique.

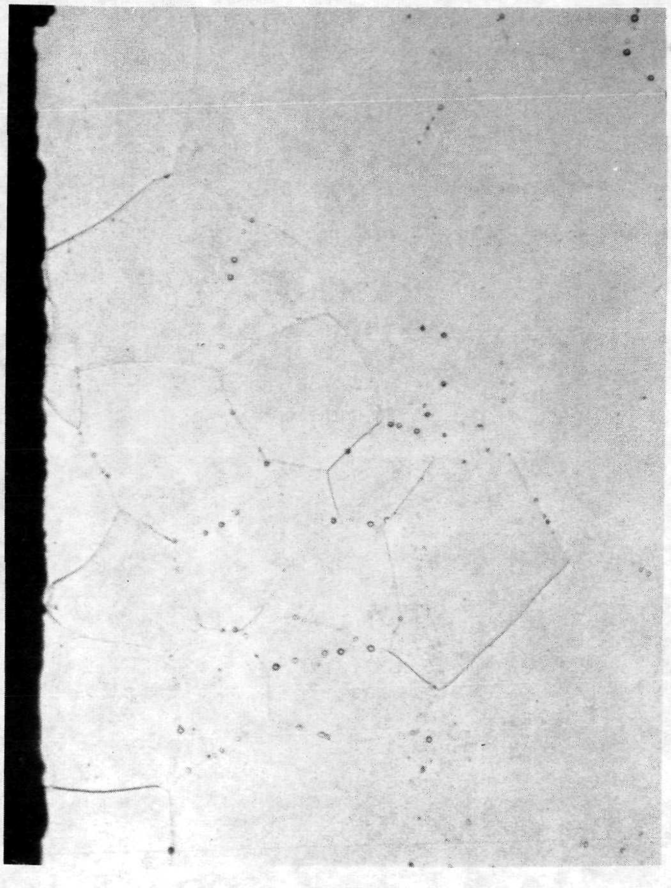
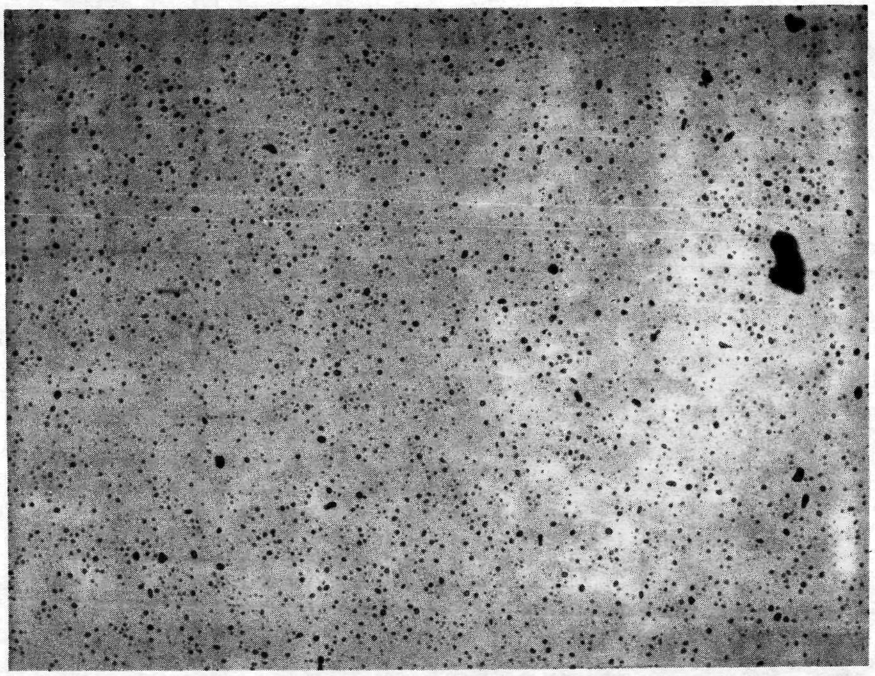
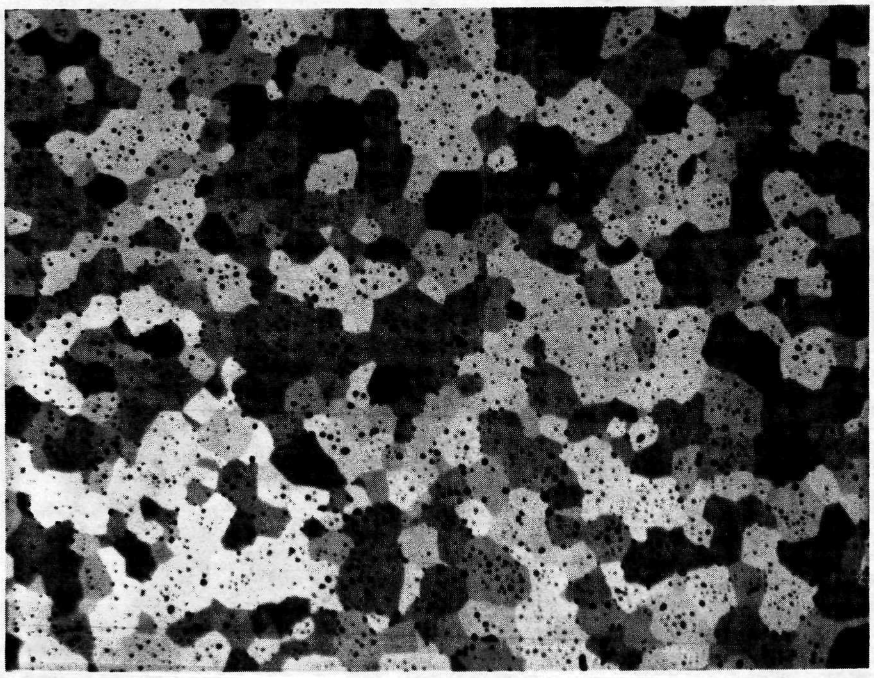


Figure 16. - Microstructure of unirradiated T-111 tubing for half-diameter fuel pin. Heated for 2862 hours at 1038° C (1900° F). Etched. X500. Etchant for all T-111 clad in this report: 30 grams of ammonium fluoride, 50 cubic centimeters of nitric acid, and 20 cubic centimeters of water.



(a) Unetched.



(b) Etched.

Figure 17. - Unirradiated uranium nitride fuel. X150. Etchant for all uranium nitride fuel in this report: 60 cubic centimeters of lactic acid, 24 cubic centimeters of nitric acid, and 2 cubic centimeters of hydrofluoric acid.

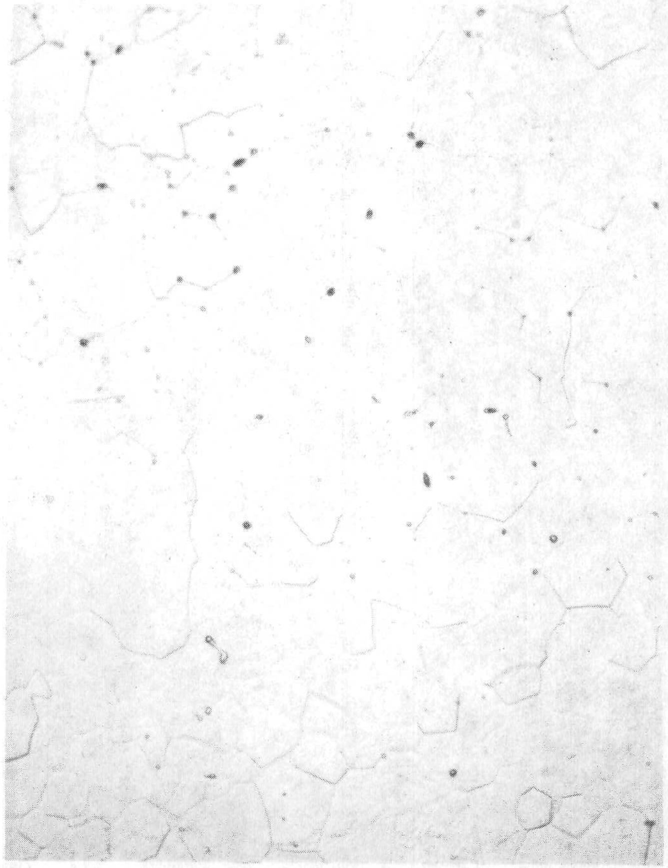
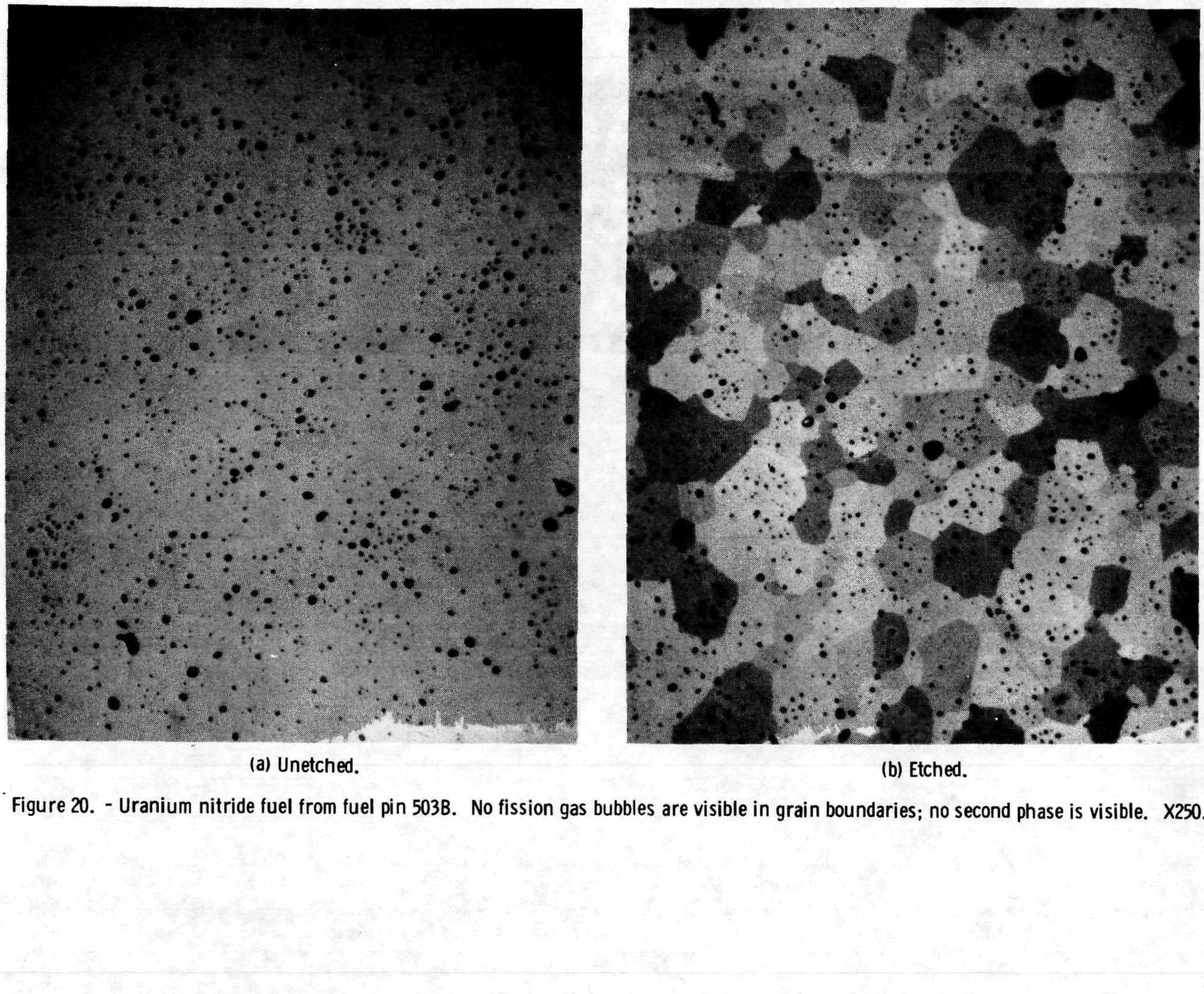


Figure 18. - Top end cap material, (T-111) of fuel pin 503B. Etched. X250.

Figure 19. - Portion of top end cap material, (T-111) between closure weld hole and thermocouple well of fuel pin 503B. Etched. X50.



(a) Unetched. (b) Etched.

Figure 20. - Uranium nitride fuel from fuel pin 503B. No fission gas bubbles are visible in grain boundaries; no second phase is visible. X250.

(a) Unetched.

(b) Etched.

Figure 21. - Uranium nitride fuel from fuel pin 503C. No fission gas bubbles are visible in grain boundaries; no second phase is visible. X250.

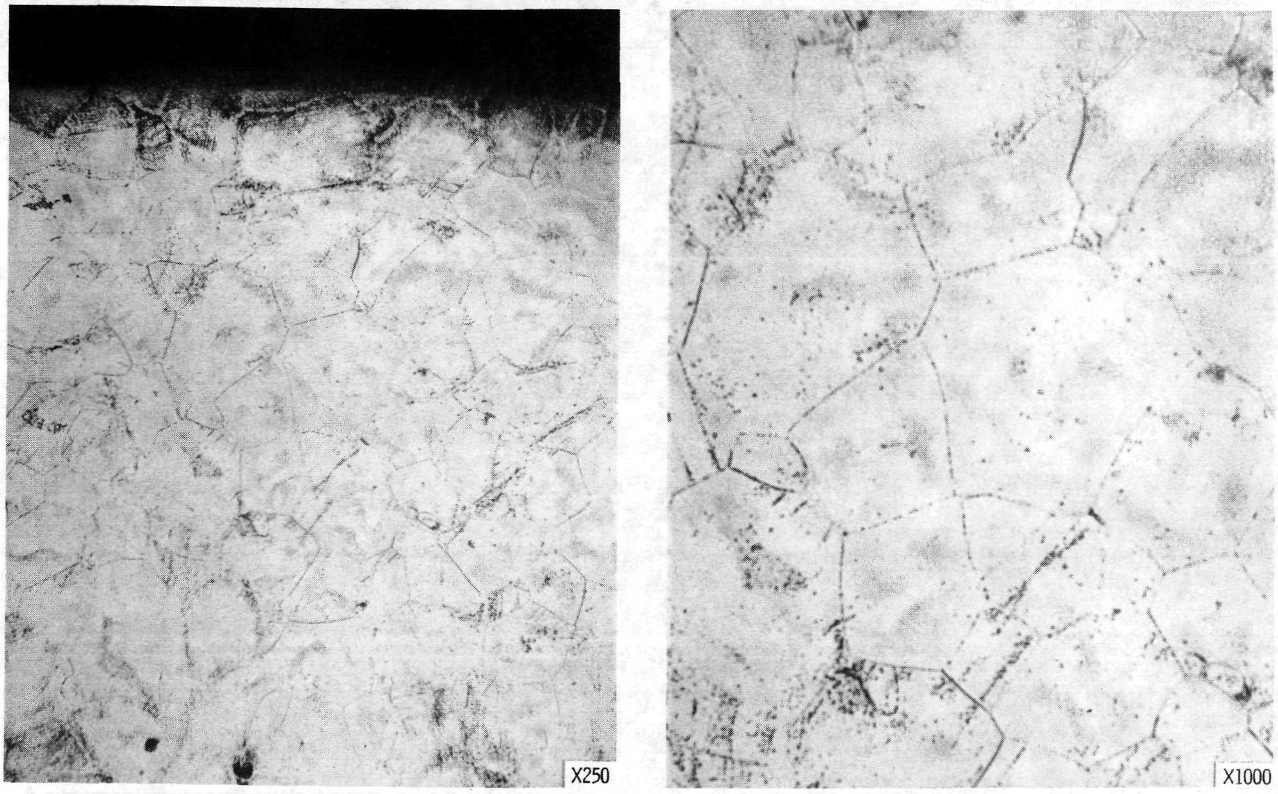


Figure 22. - T-111 cladding sample from fuel pin 503C showing heavy precipitation in metal. Etched.

Figure 23. - Top T-111 cladding of fuel pin 503C near end cap. No precipitates are visible. Etched. X250.

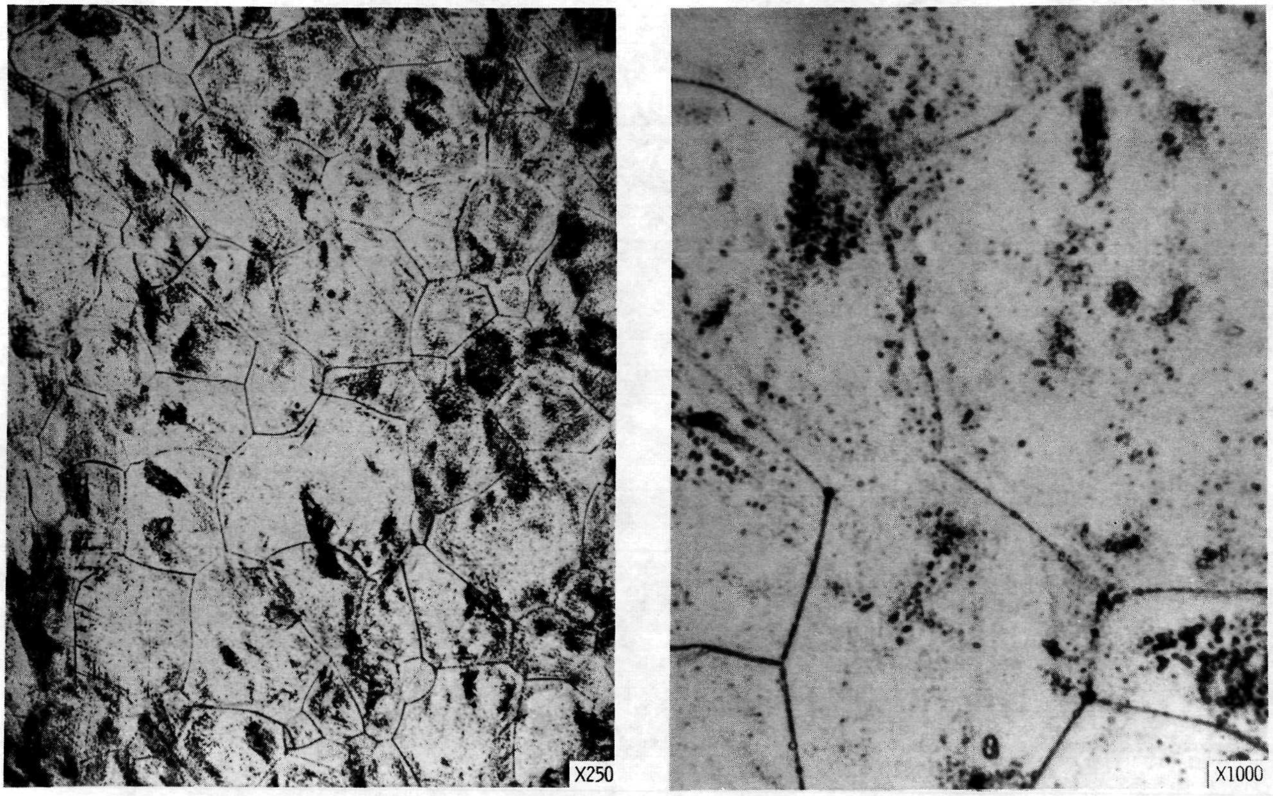


Figure 24. - T-111 cladding from fuel pin 504B showing heavy precipitation in grains and grain boundaries. Etched.

NASA-Langley, 1973 —— 22 E-7579

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